ENCLOSURE 1

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Markup of Section 1-6 of GESTAR II US Supplement

Non-Proprietary Information

S.1 Introduction

This supplement to the GESTAR II base document (Reference S–1) provides the safety analyses methodology and information specific to the GE boiling water reactor plants in the United States. A list of these plants with their associated reactor power, total number of fuel bundles, active fuel length, power density and the lattice type used in each reactor is given in Reference S–2.

Cycle–specific information for each plant reload is provided to the utility using the format given in Appendix A. No other plant–unique information is provided unless a portion of the reload does not conform to the generic document. Any deviation from the generic document will be designated in the supplemental reload licensing report and detailed in an Appendix or in a separate, referenced report to the submittal. The supplemental reload licensing report documents the number and designation of new and irradiated bundles. Plant– and cycle–specific information for initial cores is provided in the plant–specific FSAR.

Limits on plant operation are established to assure that the plant can be operated safely and not pose any undue risk to the health and safety of the public. This is accomplished by demonstrating that radioactive release from plants for normal operation, anticipated operational occurrences (AOOs) and postulated accidents meet applicable regulations in which conservative limits are documented. This conservatism is augmented by using conservative evaluation models and observing limits that are more restrictive than those documented in the applicable regulations.

Those AOOs which result in a significant reduction in MCPR or a large increase in the local power and the limiting accidents are described in this supplement along with other U.S. specific requirements as summarized in the following sections.

S.1.1 Analysis of Anticipated Operational Occurrences (AOOs) and Accidents

The effects of various postulated AOOs and accident events are investigated for a variety of plant conditions in Section S.2. The events have been categorized into three groups according to frequency of occurrence:

(1) Incidents of moderate frequency (anticipated operational occurrences).

(2) Infrequent incidents (unexpected operational occurrences).

(3) Accidents (limiting faults).

Only those events in category (1) are required to meet the design requirements for AOOs specified in Sections 2 and 4 of the GESTAR II base document (Reference S–1). Details on each of the three categories are discussed further in Section S.2.1. Descriptions of each of the significant AOO and accident events are discussed in Section S.2.2. The initial conditions and inputs to the analysis models for calculating the AOO events are discussed in Section S.2.3.

S.1.2 Vessel Pressure ASME Code Compliance

The ASME Boiler and Pressure Vessel Code and other codes and standards require that the pressure relief system prevent excessive overpressurization of the primary system process barrier and the pressure vessel. The allowable pressure and prescribed evaluations are determined by these requirements. The analysis performed to demonstrate conformance to the requirements is documented in Section S.3.

S.1.3 Stability Analysis

Stability requirements are set forth in 10CFR50 Appendix A, General Design Criterion (GDC). GDC 10 states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated occurrences. GDC 12 states that power oscillations that can result in conditions exceeding specified acceptable fuel design limits either should not be possible, or can be reliably and readily detected and suppressed.

All US BWRs have selected one of the NRC approved BWROG long–term stability solutions described in References S–79 and S–90 to meet the GDC criteria. Long–term solutions are of the prevention type (i.e., power oscillations are not possible), of the detect and suppress type (i.e., power oscillations can be reliably and readily detected and suppressed), or are a combination of the two types. Stability compliance with GDC 10 and GDC 12 must be demonstrated on a plant and cycle–specific basis for each of the long–term solutions. Stability compliance of GE BWR fuel designs is demonstrated on a generic basis only to provide assurance that plant and cycle–specific stability compliance will be provided under the applicable long–term solution.

If the long–term solution is declared inoperable due to Part 21 issues or hardware failures, the Interim Corrective Action (ICA) as outlined in Reference S–91 or an equivalent solution (e.g., the Backup Stability Protection (BSP) as outlined in Reference S–90) can be used on an interim basis. These are described in Section S.4.2.

The generic stability calculation and methodology are described in Section S.4. The plant and cycle–specific calculations required for each long–term stability solution are described in Section S.4.1.

S.1.4 Analysis Options

Several analysis options are available, on a commercial basis, to all owners of BWRs fueled by GE. As these options are selected by the BWR owners, plant–specific and/or generic–bounding analyses will be submitted for NRC approval. The first set of options provides MCPR margin improvement. The second set of options provides additional operating flexibility for BWRs. In some cases, these options are included only to describe their impact on the reload license, and separate approval must be obtained before they can be used on a specific plant. The currently available options are discussed in Section S.5.

S.2 AOO and Accident Analysis

AOOs and accident events are divided among eight individual categories in the FSARs as required by Reference S–4. The categories are as follows.

(1) **Decrease in Core Coolant Temperature**: Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity. This could lead to fuel–cladding damage.

(2) **Increase in Reactor Pressure**: Nuclear system pressure increases threaten to rupture the reactor coolant pressure boundary (RCPB). Increasing pressure also collapses the voids in the core–moderator, thereby increasing core reactivity. This could lead to fuel cladding damage.

(3) **Decrease in Reactor Core Coolant Flow Rate**: A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

(4) **Reactivity and Power Distribution Anomalies**: AOO events included in this category are those which cause rapid increases in power that are due to increased core flow disturbance events. Increased core flow reduces the void content of the moderator, thereby increasing core reactivity and power level.

(5) **Increase in Reactor Coolant Inventory**: Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc.

(6) **Decrease in Reactor Coolant Inventory**: Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.

(7) **Radioactive Release from a Subsystem or Component**: Loss of integrity of a radioactive containment component is postulated.

(8) **Anticipated Transients Without Scram**: In order to determine the capability of plant design to accommodate an extremely low probability event, a multi–system mal–operation situation is postulated.

All of the AOO and accident descriptions and analyses for initial cores are given in the plant FSAR. The purpose of this section is to discuss the significant AOOs and accidents for both initial and reload cores and classify them according to expected frequency of occurrence.

S.2.1 Frequency Classification

Each of the significant accidents and AOOs is assigned to one of the frequency groups outlined below. The frequency of occurrence of each event is summarized based upon the nuclear safety operational analysis and currently available operating plant history. The frequency classifications are as follows:

(1) **Incidents of moderate frequency** – These are incidents that may occur with a frequency greater than once per 20 years for a particular plant. This event is referred to as an "anticipated (expected) operational occurrence."

(2) **Infrequent incidents** – These are incidents that may occur during the life of the particular plant (spanning once in 20 years to once in 100 years). This event is referred to as an "abnormal (unexpected) operational occurrence."

(3) **Limiting faults** – These are incidents that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. This event is referred to as a "design basis (postulated) accident."

S.2.1.1 Unacceptable Results for Incidents of Moderate Frequency

The following are considered to be unacceptable safety results for core–wide incidents of moderate frequency (AOOs):

(1) a release of radioactive material to the environs that exceeds the limits of 10CFR20;

(2) a reactor operation induced fuel cladding failure;

(3) nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes; and

(4) containment stresses in excess of that allowed for the AOO classification by applicable industry codes.

Compliance to the above related fuel criteria (1) and (2) is conservatively demonstrated by conformance to the fuel design limits specified in Section 2 of the base document and by maintaining the MCPR above the Fuel Cladding Integrity Safety Limit MCPR identified in Reference S–2.

S.2.1.2 Unacceptable Results for Infrequent Incidents (Unexpected Operational Occurrences)

The following are considered to be unacceptable safety results for infrequent incidents (unexpected operational occurrences).

(1) release of radioactivity which results in dose consequences that exceed a small fraction (10%) of 10CFR100 (or 10% of 10CFR50.67 for Alternate Source Term plants);

(2) failure of fuel cladding which could cause changes in core geometry such that core cooling would be inhibited;

(3) generation of a condition that results in consequential loss of function of the reactor coolant system;

(4) generation of a condition that results in a consequential loss of function of a necessary containment barrier; and

(5) nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes.

S.2.1.3 Unacceptable Results for Limiting Fault (Design Basis Accidents)

The following are considered to be unacceptable safety results for limiting faults (design basis accidents):

(1) radioactive material release which results in dose consequences that exceed the guideline values of 10CFR100;

(2) failure of fuel cladding which could cause changes in core geometry such that core cooling would be inhibited;

(3) nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes;

(4) containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required; and

(5) radiation exposure to plant operations personnel in the main control room in excess of 5 Rem whole body, 30 Rem inhalation and 75 Rem skin.

S.2.2 Descriptions and Frequency Categorization of Significant AOOs, Infrequent Incidents, and Accidents

S.2.2.1 Anticipated Operational Occurrences (Moderate Frequency Events)

To determine the limiting AOO events, the relative dependency of CPR upon various thermal–hydraulic parameters was examined. A sensitivity study was performed to determine the effect of changes in bundle power, bundle flow, subcooling, R–factor and pressure on CPR for fuel designs.

Results of the study are given in Table S–1. As can be seen from this table, CPR is most dependent on the R–factor and bundle power. A slight sensitivity to pressure and flow changes and relative independence to changes in inlet subcooling was also shown. The R–factor is a function of bundle geometry and local power distribution and is assumed to be constant throughout a transient. Therefore, AOOs that would be limiting because of MCPR would primarily involve significant changes in power. Based on this, the AOOs most likely to limit operation because of MCPR considerations are:

(1) generator load rejection without bypass or turbine trip without bypass;

(2) loss of feedwater heating or inadvertent HPCI startup;

(3) control rod withdrawal error;

(4) feedwater controller failure (maximum demand); and

(5) pressure regulator downscale failure (BWR/6 only).

Subsequent AOO analyses verified the results of the above sensitivity study. Descriptions of the typical analyses performed for the above limiting events are given below. For reloads, the potentially limiting events are evaluated to determine the required operating limits. The analytical results for the limiting AOOs and the required operating limits are provided in the plant supplemental reload licensing report.

Two additional fuel loading error conditions, the mislocated bundle and the misoriented bundle event, are evaluated as infrequent incidents. If the applicability requirements in Section S.5.3 for treating the fuel loading error as an infrequent incident cannot be met, then it will be evaluated to meet the fuel cladding integrity safety limit MCPR. Descriptions of these events are given in S.2.2.2.1 for the Infrequent Incident, and S.2.2.1.8 and .9 for the AOO.

Some plant–unique analyses will differ in certain aspects from the typical calculational procedure. These differences arise because of utility–selected margin improvement options. A description of these options and their effect upon the AOO analysis is given in Section S.5. ATWS pump trip is assumed in the analysis of those plants listed in Table S–2.

The initial MCPR assumed for AOO analyses is usually greater than or equal to the GETAB operating limit. Figure 5.2–1 in Appendix B illustrates the effect of the initial MCPR on transient ΔCPR for a typical BWR core. This figure indicates that the change in ΔCPR is approximately 0.01 for a 0.05 change in initial MCPR. Therefore, nonlimiting GETAB AOO analyses may be initiated from an MCPR below the operating limit because the higher operating limit MCPR more than offsets the increase in ΔCPR for the event. This may also be applied to limiting AOOs if the difference between the operating limit and the initial MCPR is small (0.01 or 0.02).

S.2.2.1.1 Generator Load Rejection Without Bypass

Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine control valves are required to close as rapidly as possible to prevent overspeed of the turbine generator rotor. The closing causes a sudden reduction of steam flow, which results in a nuclear system pressure increase. The reactor is scrammed by the fast closure of the turbine control valves.

Starting Conditions and Assumptions. The following plant operating conditions and assumptions form the principal bases for which reactor behavior is analyzed during a load rejection:

(1) The reactor and turbine generator are initially operating at full power when the load rejection occurs.

(2) All of the plant control systems continue normal operation.

(3) Auxiliary power is continuously supplied at rated frequency.

(4) The reactor is operating in the manual flow control mode when load rejection occurs, although the results do not differ significantly for operation in the automatic flow control mode.

(5) The turbine bypass valve system is failed in the closed position.

Event Description. Complete loss of the generator load produces the following sequence of events:

(1) The power/load unbalance device steps the load reference signal to zero and closes the turbine control valves at the earliest possible time. The turbine accelerates at a maximum rate until the valves start to close. The turbine control valves on plants with electrical hydraulic turbine control (EHC) will close at a full stroke rate of approximately 0.150 sec. The turbine control valve on plants with a mechanical hydraulic turbine control (MHC) system will have a nonlinear closure signature that is a function of the MHC settings.

(2) Reactor scram is initiated upon sensing control valve fast closure.

(3) If the pressure rises to the pressure relief setpoint, part or all of the relief valves open, discharging steam to the suppression pool.

(4) On some plants, if the pressure rises above approximately 1135 psig, a trip of the recirculation pump drive motors occurs.

Identification of Operator Actions. No restart is assumed and the reactor is to be cooled down.

The operator should take the following actions:

(1) Control the reactor pressure.

(2) Ascertain that all control rods are in and that recirculation flow is at minimum.

(3) Put the reactor mode switch in the startup position before the reactor pressure decays to ≤ 850 psig.

(4) Secure the RCIC or emergency condenser if feedwater pumps are available.

(5) Check the necessity of starting the residual heat removal (RHR) system.

(6) Maintain turbine seals and steam jet air ejector (SJAE) operation.

(7) Check the turbine coastdown.

(8) When the reactor pressure decays to less than 300 psig, maintain the reactor water level using the condensate pump only, and continue steaming to the seals and SJAE until the shutdown cooling system is put into service.

(9) When the reactor is depressurized, close the main steam isolation valves (MSIVs) for maintenance on the bypass valves.

(10) Monitor torus temperature and take appropriate actions as described in the Technical Specifications.

Results and Consequences. For initial cores, the generator load rejection without bypass event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. Plant/cycle–specific results are determined using either the GENESIS, GEMINI or TRACG methods described in Subsection 4.3.1 of the GESTAR base document (Reference S–1) for reload cores.

S.2.2.1.2 Turbine Trip Without Bypass

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are: moisture separator and heater drain tank high levels, large vibrations, loss of control fluid pressure, low condenser vacuum and reactor high water level. The turbine stop valve closes, causing a sudden reduction in steam flow that results in a nuclear system pressure increase and the shutdown of the reactor.

Starting Conditions and Assumptions. The plant operating conditions and assumptions are identical to those of the generator load rejection.

Event Description. The sequence of events for a turbine trip is similar to those for a generator load rejection. Stop valve closure occurs over a typical period of 0.10 second.

Position switches at the stop valves sense the turbine trip and initiate reactor scram. If the pressure rises to the pressure relief setpoint, relief valves open, discharging steam to the suppression pool.

Identification of Operator Actions. Key operator actions required following the turbine trip without bypass are the same as required following a generator load rejection without bypass.

Results and Consequences. For initial cores, the turbine trip without bypass event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. Plant/cycle–specific results are determined using either the GENESIS, GEMINI or TRACG methods described in Subsection 4.3.1 of the GESTAR base document (Reference S–1).

S.2.2.1.3 Loss of Feedwater Heating

A loss of feedwater heating event results in a core power increase due to the increase in core inlet subcooling.

Starting Conditions and Assumptions. The following plant operating conditions and assumptions form the principal basis for which reactor behavior is analyzed during the loss of feedwater heating transient:

(1) The plant is operating at full power.

(2) The plant is operating in the manual flow control mode. The transient is moderated by the runback in core flow if operation is in the automatic flow control mode.

Event Description. Feedwater heating can be lost in at least two ways:

(1) Steam extraction line to heater is closed.

(2) Feedwater is bypassed around heater.

The first case produces a gradual cooling of the feedwater due to the stored heat capacity of the heater. In the second case, the feedwater bypasses the heater and the change in heating occurs during the stroke time of the bypass valve (about one minute, similar to the heater time constant). In either case, the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters that can be tripped or bypassed by a single event represents the most severe transient for analysis considerations and the feedwater heaters are assumed to trip instantaneously. This event causes an increase in core inlet subcooling, which increases core power due to the negative void reactivity coefficient. In automatic recirculation flow control, some compensation of core power is realized by automatic reduction of core flow.

Identification of Operator Actions. For either case, power would increase at a very moderate rate. If power exceeded the normal power flow control line, the operator would be expected to reduce recirculation flow to return the power below its initial value, and subsequently insert control rods to return to operation within the normal power/flow range. If these steps were not done, the neutron flux could exceed the scram setpoint where a scram would occur.

Results and Consequences. For initial cores, the loss of feedwater heating event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. Plant/cycle–specific results are determined using the REDY, TRACG or 3–D Simulator model.

S.2.2.1.4 Inadvertent Start of HPCI Pump (Plants with HPCI only)

This AOO is similar to the loss of feedwater heater event. The high pressure coolant injection pump is inadvertently started and the cold water injection results in an increase in inlet subcooling and a consequent increase in power. In most cases this event is bounded by the loss of feedwater heater event (Reference S–78).

Starting Conditions and Assumptions. The plant operating conditions and assumptions are identical to those of the loss of feedwater heater.

Event Description. The HPCI introduces cold water through the feedwater sparger. The normal feedwater flow is correspondingly reduced by the water level controls. The increase in inlet subcooling due to the inadvertent HPCI start is slightly less than that produced by the loss of feedwater heater event.

Identification of Operator Actions. The operator actions would be similar to those performed for the loss of feedwater heating event. In addition, the operator should determine the reason why the HPCI flow was initiated and follow proper procedures to shut off the pumps.

Results and Consequences. For initial cores, the inadvertent start of HPCI Pump event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. *If it does, the results will be reported in the supplemental reload licensing* report. The REDY, ODYN system model or the TRACG system model may also be used to simulate this event. These models are described in Subsection 4.3.1 of the GESTAR base document (Reference S–1).

S.2.2.1.5 Rod Withdrawal Error

Starting Conditions and Assumptions. The reactor is operating at a power level above 75% of rated power at the time the control rod withdrawal error occurs. The reactor operator has followed procedures and up to the point of the withdrawal error is in a normal mode of operation (i.e., the control rod pattern, flow setpoints, etc. are all within normal operating limits). For these conditions, it is assumed that the withdrawal error occurs with the maximum worth control rod. Therefore, the maximum positive reactivity insertion will occur.

Event Description. While operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod to its rod block position. Due to the positive reactivity insertion, the core average power increases. More importantly, the local power in the vicinity of the withdrawn control rod increases and could potentially cause cladding damage due to overheating, which may accompany the occurrence of boiling transition, which is an assumed AOO failure threshold. The following list depicts the sequence of events for this AOO.

(1) Event begins, operator selects the maximum worth control rod, acknowledges any alarms and withdraws the rod at the maximum rod speed.

(2) Core average power and local power increase causing LPRM alarm.

(3) Event ends – rod block by RBM or RWL.

Identification of Operator Actions. Under most normal operating conditions, no operator action will be required, since the transient that occurs will be mild. If licensing limits are exceeded, the nearest local power range monitors (LPRMs) will detect this phenomenon and sound an alarm. The operator must acknowledge this alarm and take appropriate action to rectify the situation.

If the rod withdrawal error is severe enough, the rod block monitor (RBM) system will sound alarms, at which time the operator must acknowledge the alarms and take corrective action. Even for extremely severe conditions (i.e., for highly abnormal control rod patterns, operating conditions and assuming that the operator ignores all alarms and warnings and continues to withdraw the control rod), the RBM system will block further withdrawal of the control rod before the fuel cladding integrity safety limit is exceeded.

Results and Consequences. For BWR/3, 4 and 5 plants, the ΔCPR from a rod withdrawal error is reported for each fuel type. The value reported for a particular fuel type may be from either a plant/cycle–specific analysis or the generic bounding analysis. The rod withdrawal error has been analyzed generically for BWR/6's in Reference S–5 or may be analyzed on a plant specific basis. The applicability of these generic analyses to GE fuel designs is discussed in Reference S–2.

a. Plant/Cycle–Specific Analysis

The plant/cycle–specific analysis considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor, which is operating at rated power with a control rod pattern which results in the core being placed on thermal design limits. This condition is analyzed to ensure that the results obtained are conservative; this approach also serves to demonstrate the function of the RBM system.

Results for this worst–case condition for the reload will be given in the supplemental reload licensing report. Results for late in cycle reactivity limited control rod pattern based rod withdrawal error analyses may also be reported to provide appropriate late in cycle ΔCPRs.

b. Generic Bounding Analysis (BWR/3, 4 and 5 only)

Based on the large amount of data available from past reloads, a statistical analysis was performed to calculate generic bounding values of ΔCPR as a function of rod block monitor setpoint (Reference S–6). These values are listed in Table S–3. Interim approval of this method is provided in Reference S–7. When this basis is used, the ΔCPRs are conservative relative to the actual operating limit MCPR and are valid throughout the cycle. The applicability of the generic analysis to GE fuel designs is discussed in Reference S–2.

In cases where the generic bounding analysis results in a ΔCPR that is the limiting value for a particular fuel lattice type, a plant/cycle–specific analysis may be performed for that lattice type.

S.2.2.1.6 Feedwater Controller Failure – Maximum Demand

This event is postulated on the basis of a single failure of a control device; specifically, one that can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

Starting Conditions and Assumptions. The starting conditions and assumptions considered in this analysis are as follows:

a. Feedwater controller fails during maximum flow demand.

b. Maximum feedwater pump runout is assumed.

c. The reactor is operating in a manual flow control mode, which provides for the most severe transient.

Event Description. A feedwater controller failure during maximum demand produces the following sequence of events:

a. The reactor vessel receives an excess of feedwater flow.

b. This excess flow results in an increase in core subcooling, which results in a rise in both core power and reactor vessel water level.

c. The rise in the reactor vessel water level eventually leads to high water level turbine trip, feedwater pump trip and reactor scram trip.

Identification of Operator Actions. Under most conditions, no operator action will be required. The reactor will scram on high water level and end the transient.

Results and Consequences. The influx of excess feedwater flow results in an increase in core subcooling that reduces the void fraction and thus induces an increase in reactor power. The excess feedwater flow also results in a rise in the reactor water level, which eventually leads to high water level, reactor scram, main turbine and feedwater turbine trip and turbine bypass valves being actuated. Reactor scram trip is actuated from the main stop valve position switches for plants without high water level trip. Relief valves open as steamline pressures reach relief valve setpoints.

For initial cores, the feedwater controller failure event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results are reported in the supplemental reload licensing report. Plant/cycle–specific results are determined using either the GENESIS, GEMINI or TRACG methods described in Subsection 4.3.1 of the GESTAR base document (Reference S–1).

S.2.2.1.7 Pressure Regulator Downscale Failure (BWR/6 Plants Only)

Two identical pressure regulators are provided to maintain primary system pressure control. They independently sense pressure just upstream of the main turbine stop valves and compare it to two separate setpoints to create proportional error signals that produce each regulator output. The output of both regulators feeds into a high value gate. The regulator with the highest output controls the main turbine control valves. The lowest pressure setpoint gives the largest pressure error and, thereby, largest regulator output. The backup regulator is set 5 psi higher, giving a slightly smaller error and a slightly smaller effective output of the controller.

It is assumed, for the purpose of this AOO analysis, that a single failure occurs which causes a downscale failure of the pressure regulation demand to zero (e.g., high value gate downscale failure). Should this occur, it could cause full closure of turbine control valves, as well as inhibit steam bypass flow and thereby increase reactor power and pressure. When this occurs, reactor scram will be initiated when the high neutron flux scram setpoint is reached.

This AOO is not applicable to plants with the MEOD flexibility option (see Section S.5.2.7). The MEOD evaluation concluded that the single failure initiating this AOO was very remote and did not meet the probability requirements. The pressure control of each applicable plant is reviewed to insure that it is consistent with the MEOD basis.

Starting Conditions and Assumptions. The following plant operating conditions and assumptions form the principal bases for which reactor behavior is analyzed for this event:

(1) The reactor and turbine generator are initially operating at full power when downscale failure of the pressure regulator occurs.

(2) All of the plant control systems function normal.

(3) The reactor is operating in the manual flow control mode when load rejection occurs, although the results do not differ significantly for operation in the automatic flow control mode.

Event Description. Pressure regulation downscale failure produces the following sequence of events:

(1) A failure occurs such that the high value gate receives a zero demand signal, which initiates a turbine control valve closure.

(2) Neutron flux reaches high flux scram setpoint and initiates a reactor scram.

(3) Recirculation pump drive motors are tripped due to high dome pressure. Safety/relief valves also open due to high pressure.

(4) Vessel water level trip initiates main turbine and feedwater turbine trips.

(5) Group 1 safety/relief valves open again to relieve decay heat and then reclose.

Identification of Operator Action. The operator should:

(1) monitor that all control rods are inserted;

(2) monitor reactor water level and pressure;

(3) observe turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries);

(4) observe that the reactor pressure relief valves open at their setpoint; and

(5) monitor reactor water level and continue cooldown per the normal procedure.

Results and Consequences. For initial cores, the pressure regulator downscale failure event is calculated and the ΔCPR results reported in the plant FSAR. For reload cores, an evaluation is performed to determine if this AOO could potentially alter the previous cycle MCPR operating limit. If it does, the results will be reported in the supplemental reload licensing report. Plant/ cycle–specific results are determined using either the GENESIS, GEMINI or TRACG methods described in Subsection 4.3.1 of the GESTAR base document (Reference S–1).

S.2.2.1.8 Mislocated Bundle Event

If the mislocated bundle event cannot be evaluated as an Infrequent Incident per Section S.5.3, then the event is evaluated as an AOO for initial cores and reload cores where the resultant CPR response may establish the operating limit MCPR (OLMCPR). The evaluation of the mislocated bundle as an infrequent incident is discussed in Section S.2.2.2.1.

Starting Conditions and Assumptions. Proper location of the fuel assembly in the reactor core is monitored during fuel movements and verified by procedures during core loading. Verification procedures include inventory checks, current bundle location logs, serial number verifications and visual or photographic inspection of the loaded core. The verification procedures are designed to minimize the possibility of the occurrence of the mislocated bundle accident. Plant operation with a mislocated fuel bundle is a result of a failure in the core verification process following core fueling.

Event Description.

For Initial Cores. The initial core consists of bundle types with average enrichments in the high, medium or low range with correspondingly different gadolinia concentrations. The fuel bundle loading error involves interchanging a bundle of one enrichment with another bundle of a different enrichment. The following fuel loading errors are possible in an initial core:

(1) A high–enrichment bundle is misloaded into low–enrichment bundle location.

(2) A medium–enrichment bundle is misloaded into a low–enrichment bundle location.

(3) A low–enrichment bundle is misloaded into a high–enrichment bundle location.

(4) A low–enrichment bundle is misloaded into a medium–enrichment bundle location.

(5) A medium–enrichment bundle is misloaded into a high–enrichment bundle location.

(6) A high–enrichment bundle is misloaded into a medium–enrichment bundle location.

Because all low–enrichment bundles are located on the core periphery, the misloading of high– or medium–enrichment bundles into a low–enrichment bundle location [misloading errors (1) or (2)] is not significant. In these cases, the higher reactivity bundles are moved to a region of low reactivity and power resulting in an overall improvement in performance and no impact on thermal margin.

The third type of fuel loading error results in the largest enrichment mismatch. For initial cores using thermal traversing in–core probes (TIPs), this loading error does not result in an unacceptable operating consequence. Consider a fuel bundle loading error at beginning–of–cycle (BOC) and the low–enrichment bundle interchanged with a high–enrichment bundle located adjacent to the Local Power Range Monitor (LPRM) and predicted to be closest to technical specification limits. After the loading error has occurred and has gone undetected, assume, for purposes of conservatism, that the operator uses a control pattern that places the limiting bundle in the four–bundle array containing the misplaced bundle on thermal limits as recorded by the LPRM. As a result of loading the low–enrichment bundle in an improper location, the average power of the four bundles decreases. Normally, the reading of the LPRM will show a decrease in thermal flux due to the decreased power; however, in this case, an increase in the thermal flux occurs due to decreased neutron absorption in the low–enrichment bundle. The effect of the decreased thermal absorption is larger than the effect of power depression resulting in a net increase in the instrument reading. Thus, detected reductions in thermal margins during power operations will indicate a fuel loading error of this kind.

The fourth and fifth types of fuel loading errors are similar to the third type and also result in conservative operating errors.

The fuel bundle loading error with greatest impact on thermal margin is of the sixth type, which occurs when a high–enrichment bundle is interchanged with a medium–enrichment bundle located away from an LPRM. Since the medium– and high–enrichment bundles have corresponding medium and high gadolinia contents, the maximum reactivity difference occurs at the end of cycle (EOC) when the gadolinia has burned out.

For Reload Cores. The loading error involves the mislocation of at least two fuel bundles. One location is loaded with a bundle that would potentially operate at a lower critical power than it would otherwise. The other location would operate at a higher critical power. The low critical power location could have less margin to boiling transition than other bundles in the core; therefore, the MCPR operating limit is set to protect against this occurrence.

Identification of Operator Actions. There is a possibility that core monitoring will provide information that allows the operator or reactor engineer to recognize that an error exists and determine appropriate mitigating actions. Where the high radial power mislocated bundle is adjacent to an instrument, the power adjustment in radially TIP or LPRM adapting monitoring systems will cause higher monitored bundle power. The reactor will be operated such that the most limiting of the bundles near the mislocation will be maintained below the operating limit MCPR. Where the mislocated bundle has a bundle between it and the instrument, the core monitoring may not recognize the mislocation.

If loading errors were made and have gone undetected, the operator would assume that the mislocated bundle would operate at the same power as the instrumented bundle in the mirror–image location and would operate the plant until EOC. For the purpose of conservatism, it is assumed that the mirror–image bundle is on thermal limits as recorded by the LPRM. As a result of placing the instrumented bundle on limits, the mislocated bundle may violate the Tech Spec operating limit MCPR.

Results and Consequences. Assuming the mislocated bundle is not identified, it is possible that the fuel bundle operates through the cycle close to or above the fuel thermal mechanical limit. Therefore, the MCPR operating limit is set to protect against this occurrence.

Further discussion on the analysis methods for the mislocated bundle accident is given in References S–45 and S–46.

S.2.2.1.9 Misoriented Bundle Event

If the misoriented bundle event cannot be evaluated as an Infrequent Incident per Section S.5.3, then the event is evaluated as an AOO for initial cores and reload cores where the resultant CPR response may establish the operating limit MCPR (OLMCPR). The evaluation of the misoriented bundle as an infrequent incident is discussed in Section S.2.2.2.1.

Starting Conditions and Assumptions. Proper orientation of the fuel assembly in the reactor core is monitored during fuel movements and verified by procedures during core loading. Five separate visual indications of proper fuel assembly orientation exist:

(1) The channel fastener assemblies, including the spring and guard used to maintain clearances between channels, are located at one corner of each fuel assembly adjacent to the center of the control rod.

(2) The identification boss on the fuel assembly handle points toward the adjacent control rod.

(3) The channel spacing buttons are adjacent to the control rod passage area.

(4) The assembly identification numbers that are located on the fuel assembly handles are all readable from the direction of the center of the cell.

(5) There is cell–to–cell replication.

Event Description. This fuel loading error involves the misorientation of a single fuel bundle. The power distribution in the misoriented bundle would be affected as well as its neighbors. The resulting power distribution could reduce the margin to boiling transition.

Identification of Operator Actions. There is a possibility that core monitoring will provide information that allows the operator or reactor engineer to recognize that an error exists and determine appropriate mitigating actions. If loading errors were made and have gone undetected, the plant would continue to operate until EOC.

Results and Consequences. Assuming the mislocated bundle is not identified, it is possible that the fuel bundle operates through the cycle close to or above the fuel thermal mechanical limit. Therefore, the MCPR operating limit is set to protect against this occurrence.

Analysis methods for the misoriented fuel assembly are discussed in detail in Reference S–46. Approval of these methods is given in Reference S–47 under the stipulation that a ΔCPR penalty of 0.02 be added for the tilted misoriented bundle. This 0.02 is added on to the calculated ΔCPR used in determining the operating limit when utilizing this method. GE applies the Fuel Cladding Integrity Safety Limit discussed in Section 4 of the base GESTAR II document (Reference S–1) and presented in Reference S–2 to the accident results reported in the plant FSAR or the supplemental reload licensing report. Individual utilities may elect to substitute an alternative approach as noted in Reference S–47 for this limit.

S.2.2.2 Unexpected Operational Occurrences (Infrequent Incidents)

S.2.2.2.1 Fuel Loading Error (Mislocated or Misoriented Bundle Event)

A generic bounding analysis of the fuel loading error (mislocated or misoriented bundle event) is provided in Reference S-99. The plant must meet the requirements of Section S.5.3 in order to apply this generic analysis. If the plant cannot meet the requirements of S.5.3, then the mislocated or misoriented bundle is evaluated as discussed in Sections S.2.2.1.8 and .9.

Starting Conditions and Assumptions. Proper location and orientation of the fuel assemblies in the reactor core is monitored during fuel movements and verified by procedures during core loading. Verification procedures address location, orientation, and seating through visual examinations of the loaded core. The verification procedures are designed to minimize the possibility of the occurrence of the mislocated or misoriented bundle event. Plant operation with a mislocated or misoriented fuel bundle is a result of a failure in the core verification process following core fueling.

Event Description. The description of the mislocated or misoriented fuel bundle event is the same as in Sections S.2.2.1.8 and .9 except that for the infrequent incident it is assumed that the event proceeds to cause fuel failures.

Identification of Operator Actions. The initial core power distribution indications and possible operator actions are the same as in Sections S.2.2.1.8 and .9. Should fuel failures occur, the offgas activity quickly increases. At that point, the operator would take steps to reduce power or scram the reactor to reduce or terminate the release.

Results and Consequences. Reference S-99 provides a bounding analysis based on a very conservative assumption of all of the fuel rods failing in five fuel bundles. Two scenarios for the fuel loading error were considered. The first assumed that the fission product activity is airborne in the turbine and condenser following Main Steam Isolation Valve (MSIV) closure and leaks directly from the condenser to the atmosphere. In the second scenario, it was assumed that no automatic MSIV closure occurred and that the activity was transported to an augmented offgas system. Calculations of post–accident doses for the Exclusion Area Boundary (EAB) were performed for each scenario to compare radiological consequences with the applicable exposure limits. EAB doses were also calculated for both scenarios utilizing the alternate source term methodology.

The plant specific offgas system parameters and site atmospheric dispersion parameters are used to confirm the applicability of the EAB generic analysis. A conservative analysis for the control room dose was also established such that plant specific atmospheric dispersion parameters can also be used to confirm its applicability. Section S.5.3 defines the items that must be confirmed and documented with the reload design documentation to support application of the Infrequent Incident analysis option.

S.2.2.3 Design Basis Accidents (Limiting Faults)

In this category, evaluations of less frequent postulated events are made to assure an even greater depth of safety. Accidents are events that have a projected frequency of occurrence of less than once in every one hundred years for every operating BWR. The broad spectrum of postulated accidents is covered by five categories of design basis events. These events are the control rod drop, loss–of–coolant, main steam line break, one recirculation pump seizure, and refueling accident.

S.2.2.3.1 Control Rod Drop Accident Evaluation

There are many ways of inserting reactivity into a boiling water reactor; however, most of them result in a relatively slow rate of reactivity insertion and therefore pose no threat to the system. It is possible, however, that a rapid removal of a high worth control rod could result in a potentially significant excursion; therefore, the accident that has been chosen to encompass the consequences of a reactivity excursion is the control rod drop accident (RDA). The dropping of the rod results in a high local reactivity in a small region of the core and for large, loosely coupled cores, significant shifts in the spatial power generation during the course of the excursion.

S.2.2.3.1.1 Sequence of Events

The sequence of events and approximate time of occurrence for this accident are described below:

	Banked Position Withdrawal Sequence (BPWS) Plants — Event	Approximate Elapsed Time
(a)	Reactor is at a control rod pattern corresponding to maximum increment rod worth.	–
(b)	Rod pattern control systems (Rod Worth Minimizer, Rod Sequence Control System, or Rod Pattern Controller) or operators are functioning within constraints of BPWS. The control rod that will result in the maximum incremental reactivity worth addition at any time in core life under any operating condition while employing the BPWS becomes decoupled from the control rod drive.	–
(c)	Operator selects and withdraws the drive of the decoupled rod along with the other required control rods assigned to the Banked–position group such that the proper core geometry for the maximum incremental rod worth exists.	–
(d)	Decoupled control rod sticks in the fully inserted position.	–
(e)	Control rod becomes unstuck and drops at the maximum velocity determined from experimental data (3.11 fps).	0
(f)	Reactor goes on a positive period and initial power burst is terminated by the Doppler reactivity feedback.	≤ 1 sec.
(g)	APRM 120% power signal scrams reactor (conservative; in startup mode APRM scram would be operative + IRM).	–
(h)	Scram terminates accident.	≤ 5 sec.

(1) **Banked Position Withdrawal Sequence (BPWS) Plants:**

All plants except those listed in Table S–4 utilize the BPWS. Those Group Notch plants in Table S–4 that have modified their Rod Worth Minimizer (RWM) and have provided a separate submittal to the NRC, which enforces the BPWS as described in Reference S–9, are included in this group.

Those plants listed in Table S–4 that have implemented the modifications described in Reference S–10 are also included in this group. Plants that implement the modifications described in Reference S–10 must modify their technical specifications to assure high operability of their rod pattern control system, review procedures and quality control for second operator substitution, and provide a discussion of this review to the NRC.

To limit the worth of the rod that would be dropped in a BPWS plant, the rod pattern control systems are used below the plant–specific low power setpoint to enforce the rod withdrawal sequence. These systems are programmed to follow the bank position withdrawal sequences (BPWS), which are generically defined in

Reference S–11. Plants that have implemented the BPWS in accordance with Reference S-11 may also implement the Improved BPWS Control Rod Insertion Process as defined in Reference S-101.

	Group Notch Plants — Event	Approximate Elapsed Time
(a)	Reactor is at a control rod pattern corresponding to maximum increment rod worth.	–
(b)	Rod worth minimizer is not functioning. Maximum worth control blade that can be developed at any time in core life under any operating conditions with the group notch RSCS operational becomes decoupled from the control rod drive.	–
(c)	Operator selects and withdraws the control rod drive of the decoupled maximum worth rod along with the other required control rods assigned to its Rod Sequence Control System group such that the proper core geometry for the maximum incremental rod worth exists.	–
(d)	Decoupled control rod sticks in the fully inserted position.	–
(e)	Blade becomes unstuck and drops at the maximum velocity determined from experimental data (3.11 fps).	0
(f)	Reactor goes prompt critical and initial power burst is terminated by the Doppler reactivity feedback.	≤ 1 sec.
(g)	APRM 120% power signal scrams reactor (conservative; in startup mode APRM scram would be operative + IRM).	–
(h)	Scram terminates accident.	≤ 5 sec.

(2) **Group Notch Plants**

Plants listed in Table S–4 are Group Notch plants. Those Group Notch plants that enforce the BPWS as described in Reference S–9, are not included in this group. In addition, those plants that have implemented the modifications of Reference S–10 are also not included in this group.

To limit the worth of the rod that could be dropped in a group notch plant that has not implemented the modifications of Reference S–10, a group notch rod sequence control system (RSCS) is installed to control the sequence of rod withdrawal. This system prevents the movement of an out–of–sequence rod before the 50% rod density configuration is achieved (except for plants operating in the BPWS mode described in Reference S–9), and prevents high–control rod worth beyond the 50% rod density configuration by enforcing a group notch mode of rod withdrawal. The 50% rod density configuration occurs during each reactor startup and corresponds to a "checkerboard" rod pattern in which 50% of the rods are fully inserted in the core and 50% are fully withdrawn. The rod drop accident design limit restricts peak enthalpies in excess of 280 cal/gm for any possible plant operation or core exposure.

S.2.2.3.1.2 Analytical Methods.

Techniques and models used to analyze the rod drop accident (RDA) are documented in References S–12, S–13, S–14 and S–9. The information in these documents has been used for the development of design approaches to make the consequences of RDA acceptable.

(1) Banked Position Withdrawal Sequence (BPWS) Plants

Control rod drop accident (CRDA) results from BPWS plants have been statistically analyzed and documented in Reference S–15. The results show that, in all cases, the peak fuel enthalpy in an RDA would be much less than the 280–cal/gm design limit even with a maximum incremental rod worth corresponding to 95% probability at the 95% confidence level. Based on these results, it was proposed to the NRC, and subsequently found acceptable, to delete the CRDA from the standard GE BWR reload package for the BPWS plants.

Because of the large margin available to CRDA design limits for BPWS plants, implementation of the advanced physics methods (Reference S–16) does not result in challenging the 280–cal/gm limit. Therefore, the impact of using the advanced physics methods of Reference S–16 as compared to the physics methods described in Reference S–17 on the generic BPWS analysis, is considered negligible. Applicability of the generic BPWS analysis to GE fuel designs is given in Reference S–2.

(2) Group Notch Plants

For group notch plants not operating in the BPWS mode described in Reference S–9 or that have not implemented the modifications described in Reference S–10, the highest control rod worth in the cold condition is determined for a series of rod drop states. Hot–standby cases are also run for any cold case that is not subcritical. The resultant peak fuel enthalpy for cold and, if applicable, hot–standby is then determined. This enthalpy value is then compared to the 280 cal/gm RDA design limit. The CRDA calculational procedures are independent of whether the physics models of either Reference S–17 or Reference S–16 are used.

Group notch plants operating in the BPWS mode described in Reference S–9 or those plants that have implemented the modifications of Reference S–10 can reference the statistical CRDA analysis documented in Reference S–15. This will allow these plants to delete the CRDA analysis from the standard GE–BWR reload package.

Results of the analysis for reload cores are supplied in the specific plant supplemental reload licensing report. For those group notch plants not operating in the BPWS mode described in Reference S–9, these results include the resultant peak enthalpy in the cold and, if applicable, the hot–standby condition.

S.2.2.3.1.3 Effect of Fuel Densification

The effect of axial gap formation due to fuel densification on the rod drop accident results is discussed in Reference S–18. Based on this evaluation, it has been established that there is a 99% probability that increased local peaking in any fuel rod due to the formation of axial gaps will be less than 5%. This effect has been accommodated by adjusting the local peaking factor.

S.2.2.3.1.4 Results and Consequences

Results of radiological analyses for initial cores are reported in the FSAR. For reloads, based on a bounding analysis, it was conservatively determined that 850[1] fuel rods would reach a fuel enthalpy of 170 cal/gm, which is the enthalpy limit for eventual cladding perforation. Safety analysis reports written prior to the development of the model and techniques reported previously, and those used to predict the 850 failures, resulted in the failure of approximately 330 fuel rods for the 7x7 fuel. Based on these new models and assumptions, the resultant number of failures for a 7x7 core would be 660 fuel rods. If the conservative assumption is made that the fractional plenum activity for 8x8, 8x8R, P8x8R, and BP8x8R fuel is the same as for the 7x7 fuel, the resultant increase in activity released from the 8x8 fuel and the subsequent radiological exposures relative to 7x7 analysis for the failure of 330 rods is (850/330) (49/63) = 2 times the 7x7 analysis. As noted in the FSAR, even if the radiological exposures are increased by a factor of two, the effects are still orders of magnitude below those identified in 10CFR100. The radiological consequences of the CRDA, assuming a full core of more recent GE fuel designs, are discussed in Reference S–2.

Results of the enthalpy analysis for initial cores are reported in the FSAR.

Results of the analysis for reload cores are supplied in the specific plant supplemental reload licensing report. For group notch plants that are not operating in the BPWS mode described in Reference S–9 or that have not implemented the modifications of Reference S–10, these results include the resultant peak enthalpy in the cold and, if applicable, the hot–standby condition.

S.2.2.3.2 Loss–of–Coolant Accident

Two separate emergency core cooling system (ECCS) evaluation methodologies are available to determine the effects of the loss–of–coolant accident (LOCA) in accordance with the requirements of 10CFR50.46 and Appendix K. Either methodology can be used to calculate the LOCA results. The particular method used is the utility's option and depends upon economic and not safety considerations. The method used will be indicated in the FSAR for initial cores or the supplemental reload licensing report for each cycle (see Appendix A of country–specific supplement).

[1] Includes a 10% allowance for uncertainties in the calculation.

The first methodology (SAFE/REFLOOD), identified in Sections S.2.2.3.2.1 and S.2.2.3.2.3 and discussed in detail in Reference S–19, utilizes conservative thermal–hydraulic/heat transfer correlations and conservative bounding values for key inputs. The resulting calculated peak cladding temperature (PCT) consists of compounded conservatisms and therefore is unrealistically high. However, as long as the resultant PCT is less than 2200°F (10CFR50.46 limit) and plant operation is not unduly restricted in order to remain under that limit, then this conservative method may satisfy utility needs.

The second methodology (SAFER/GESTR), identified in Sections S.2.2.3.2.4 and S.2.2.3.2.5, utilizes improved ECCS evaluation models (References S–20 and S–21) along with a realistic application approach (Reference S–22) to calculate a licensing PCT with margin substantiated by statistical considerations. Nominal values are used for most inputs, and Appendix K required inputs are utilized only for the limiting break in order to establish a licensing margin to 10CFR50.46 limits. This methodology was revised in Reference S–74 to extend the application to non–jet pump plants. Use of this improved methodology is optional and is dependent upon economic benefits and not safety concerns.

S.2.2.3.2.1 SAFE/REFLOOD LOCA Model Descriptions

Five different GE computer models are utilized to calculate LOCA analysis results for a BWR. Conservative values are used along with required Appendix K criteria as input to these models. The models are summarized below and discussed in detail in Reference S–19. NRC approval of this LOCA model and calculational procedure is given in Reference S–23. These models are applicable to prepressurized fuel and have been approved for prepressurized fuel in Reference S–24. Non–prepressurized fuel calculations result in conservative limits with respect to prepressurized fuel. The MAPLHGR values calculated by the codes described below are applicable to both nonbarrier and barrier fuel.

S.2.2.3.2.1.1 Short–Term Thermal–Hydraulic Model (LAMB)

The LAMB code is a model that is used to analyze the short–term thermodynamic and thermal–hydraulic behavior of the coolant in the vessel during a postulated loss–of–coolant accident. In particular, LAMB predicts the core flow, core inlet enthalpy and core pressure during the blowdown prior to the end of lower plenum flashing (~20 to 40 seconds depending on break size being evaluated). For a detailed description of the model and a discussion regarding sources of input to the model, refer to the LAMB Code Documentation portion of Reference S–19.

S.2.2.3.2.1.2 Transient Boiling Transition Model (SCAT)

The SCAT model is used to evaluate the short–term thermal–hydraulic response of the coolant in the core during a postulated loss–of–coolant accident. In particular, the convective heat transfer process in the thermally limiting fuel bundle is analyzed during the transient. For a detailed description of the model and a discussion regarding sources of input to the model, refer to the SCAT Code Documentation portion of Reference S–19.

S.2.2.3.2.1.3 Long–Term Thermal–Hydraulic Model (SAFE/REFLOOD)

The SAFE model is used to analyze the long–term thermal–hydraulic behavior of the coolant in the vessel for all breaks. The SAFE and REFLOOD models calculate the uncovery and reflooding of the fuel and the duration of spray cooling. For a detailed description of the SAFE model and a discussion regarding sources of input to the model, refer to the SAFE Code Documentation portion of the Reference S–19.

Amendment 4, Saturated Counter–Current Flow Characteristics of a BWR Upper Tieplate, of Reference S–19 is a detailed description of the counter–current flow limiting (CCFL) of a BWR in the upper tieplate during saturated and subcooled water spray of the core. The CCFL phenomenon is modeled with a correlation based on experiments with electrically heated fuel bundles. Currently, no credit is taken for this ECCS model improvement. Not utilizing this model partially compensates for the non–conservative fission gas release correlation currently utilized with SAFE/REFLOOD (see Section S.2.2.3.2.3.4).

The REFLOOD model is used for all break sizes to calculate the system inventories after ECCS actuation when core reflooding occurs. REFLOOD accounts for the numerous bypass flow paths that exist in a BWR between the core and bypass regions. These bypass regions serve the important function of helping to refill the lower plenum and subsequently reflood the core region. For a detailed description of the REFLOOD model and a description regarding sources of input to the model, refer to the REFLOOD Code Documentation portion of Reference S–19.

S.2.2.3.2.1.4 Core Heatup Model (CHASTE)

The CHASTE model solves the transient heat transfer equations, for the highest power axial plane of the highest power assembly, for the entire LOCA transient. For a detailed description of the CHASTE model and a discussion regarding sources of input to the model, refer to the CHASTE Code Documentation section of Reference S–19.

The modified Bromley heat transfer correlation provides improved heat transfer credit for the time between departure from nucleate boiling (DNB) until the fuel rods become uncovered. This low flow film boiling period helps remove heat from the core and is described in detail in Amendment 1, Calculation of Low Flow Film Boiling Heat Transfer for BWR LOCA Analysis, of Reference S–19. As with the CCFL correlation (see Section S.2.2.3.2.1.3), no credit is taken for the Bromley model in ECCS analyses. This correlation, along with the CCFL correlation, compensates for the non–conservative fission gas release correlation currently utilized with SAFE/REFLOOD (see Section S.2.2.3.2.3.4).

The core heatup model used for the analysis is that described in Reference S–19. The model has been used to predict the results of a number of ECCS transient tests of a full–scale, stainless steel–clad heater rod bundle. These tests confirm the conservatism of the model as used for reload fuel.

The fuel rod cladding rupture temperature model, which describes the thermal–mechanical conditions that will result in fuel rod perforation, and the corresponding cladding strain model, which describes the extent of cladding deformation before and after perforation occurs, are discussed in Reference S–19. Further discussion of GE's cladding rupture and strain models, as related to NUREG–0630 requirements, is given in References S–26, S–27, S–28 and S–29. NRC approval of the rupture and strain models, as modified by these references, is given in a supplementary SER (Reference S–30).

S.2.2.3.2.2 Effect of Fuel Densification

Power spiking due to in–reactor fuel densification has not been explicitly considered in LOCA calculations submitted to the NRC. Approval of GE's analytical procedure to account for the effects of fuel densification power spiking is given in Reference S–31.

S.2.2.3.2.3 SAFE/REFLOOD LOCA Model Application Methodology

The previously described models and computer codes can be used to evaluate all plants. The LAMB Code calculates the short–term blowdown response and core flow, which are input to the SCAT code to calculate blowdown heat transfer coefficients. The SAFE code is used to determine longer–term system response and flows from the various ECC systems. Where appropriate, the output of SAFE is used in the REFLOOD code to calculate liquid levels. The results of these codes are used in the CHASTE code to calculate fuel rod cladding temperatures and maximum average planar linear heat generation rates (MAPLHGR) for each fuel type.

Most operating plants have been separated into groups for the purpose of LOCA analysis (Reference S–32). Within each plant group there will be a single lead plant analysis which provides the basis for the selection of the most limiting break size yielding the highest PCT. Also, the lead plant analysis provides an expanded documentation base to provide added insight into evaluation of the details of particular phenomena. The remainder of the plants in that group will have plant–specific analyses referenced to the lead plant analysis. The plant–specific LOCA analysis and the reference lead plant analysis for each plant is indicated in Table S–5.

Additional details of the analysis and justification for the choice of inputs for the reload analysis are given in Reference S–19. The difference in input parameters is not expected to result in significantly different results for the plants within a given group. Emergency core cooling system (ECCS) and geometric differences between plant groups may result in different responses for different groups but within any group the responses will be similar.

The LOCA analysis for each plant not specifically identified in Table S–5 is provided in the individual plant FSAR.

S.2.2.3.2.3.1 Lead Plant Selection

Lead plants are selected and analyzed in detail to permit a more comprehensive review and eliminate unnecessary calculations. This constitutes a generic analysis for each plant of that type which can be referenced in subsequent plant submittals.

Based on the criteria given in Reference S–32, the BWR/2s through BWR/4s have been divided into four groups. A lead plant was selected for each group whose LOCA response would be representative of the entire group. The four groups are identified as BWR/2, BWR/3, BWR/4 with loop selection logic (plants that have not incorporated the low pressure coolant injection (LPCI) system modification), and BWR/4 with LPCI modification.

For BWR/5 and BWR/6 plants, no lead plant was selected. Each of these plant analyses was performed on a plant–specific basis.

S.2.2.3.2.3.2 BWR/3 and BWR/4s

For BWR/3s and BWR/4s, the full complement of the LOCA codes (LAMB, SCAT, SAFE, REFLOOD, CHASTE) are used to evaluate the entire spectrum of break sizes as described in Reference S–19. These plants have been divided into three groups for the purpose of analysis: (1) BWR/3; (2) BWR/4 without LPCI modification; and (3) BWR/4 with LPCI modification. One BWR/3 is included in the second group due to similarities in bypass flow and reflooding characteristics.

Application of the LOCA analysis methods for partial and full core drilling of fuel bundles in the BWR/3s is covered in Reference S–33 and in BWR/4s in References S–34, S–35, S–36 and S–37. Approval for the LOCA analysis methods for BWR/3s is given in Reference S–38.

Application of the LOCA analysis methods in the evaluation of the effects of less than rated initial core flow is presented in Reference S–39. Approval of this evaluation is presented in Reference S–40.

S.2.2.3.2.3.3 Extension of ECCS Performance Limits

The effect of increased fission gas release from the fuel associated with higher exposures (greater than 33 GWd/MTU) on MAPLHGR has been evaluated (References S–41 and S–42). The evaluation shows that for BWR/3–6, PCT margins to the regulatory limit of 2200°F, when combined with PCT reductions due to ECCS model improvements (described in Amendments 1 and 4 of Reference S–19), will more than compensate for the PCT increase associated with increased fission gas release. Therefore, exposure–dependent fission gas release can be specifically accounted for without reducing current and proposed MAPLHGR technical specifications, provided no credit is taken for the ECCS model changes. NRC approval of this is given in Reference S–43. The impact of fission gas release will be analyzed on a case–by–case basis if the improved ECCS models are used in the ECCS performance analysis or if PCT margins are less than those specified in Reference S–42.

S.2.2.3.2.4 SAFER/GESTR LOCA Model Code Descriptions

Results of extensive LOCA experimental programs since 1974 have clearly demonstrated the large conservatisms that the SAFE/REFLOOD LOCA models (Section S.2.2.3.2.3) have with respect to modeling the vessel inventory, inventory distribution and core heat transfer. A new thermal–hydraulic model (SAFER) and a new fuel rod thermal–mechanical model (GESTR–LOCA) have been developed to provide more realistic calculations for LOCA analyses. The

SAFER and GESTR–LOCA models are summarized below and discussed in detail in References S–20, S–21, S–74, S–92 (as reviewed by the NRC in the letter specified in Reference S–92) and S–93.

As with the SAFE/REFLOOD LOCA models (Section S.2.2.3.2.1), SAFER/GESTR–LOCA is also applicable to prepressurized fuel. Non–prepressurized fuel calculations result in conservative limits with respect to prepressurized fuel. The MAPLHGR values calculated by the codes are applicable to both nonbarrier and barrier fuel.

S.2.2.3.2.4.1 Realistic Thermal–Hydraulics Model (SAFER)

SAFER replaces the combination of the SAFE and REFLOOD ECCS performance evaluation models discussed in Section S.2.2.3.2.1.3.

The SAFER code employs a heatup model with a simplified radiation heat transfer correlation to calculate PCT and local maximum oxidation, which replaces the CHASTE heatup calculation (Section 2.2.3.2.1.4). The PCT and local maximum oxidation fraction from SAFER can be used directly.

S.2.2.3.2.4.2 Best Estimate Fuel Rod Thermal Mechanical Model (GESTR–LOCA)

The GESTR–LOCA model has been developed to provide best–estimate predictions of the thermal performance of GE nuclear fuel rods experiencing variable power histories. For ECCS analyses, the GESTR–LOCA model is used to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA. Details of the GESTR–LOCA models are provided in Reference S–20.

S.2.2.3.2.4.3 Transient Boiling Transition Model (TASC)

TASC replaces the SCAT boiling transition model discussed in Section S.2.2.3.2.1.2.

The TASC model is used to evaluate the short–term thermal–hydraulic response of the coolant in the core during a postulated loss–of–coolant accident. In particular, the convective heat transfer response in the thermally limiting fuel bundle is analyzed during the transient. For a detailed description of the model and a discussion regarding sources of input to the model, refer to Reference S–94.

S.2.2.3.2.5 SAFER/GESTR–LOCA Model Application Methodology

Using the SAFER/GESTR–LOCA models, the LOCA events are analyzed with nominal values of inputs and correlations. A calculation is performed in conformance to Appendix K and checked for consistency with generic statistical upper bound analyses that encompass modeling uncertainties in SAFER/GESTR–LOCA and uncertainties related to plant parameters.

As with the SAFE/REFLOOD LOCA models application methodology (Section S.2.2.3.2.2), the effects of power spiking due to in–reactor densification are considered negligible for SAFER/GESTR–LOCA analyses for similar reasons.

The details of the application methodology are summarized below and discussed in detail in References S–22, S–92 and S–93. The plant–specific LOCA analysis report for each plant is identified in Table S–5.

S.2.2.3.2.5.1 Appendix K Conformance

The SAFER/GESTR–LOCA Appendix K conformance calculation will be performed only for the limiting break of a nominally calculated break spectrum with a range of break flow multipliers between 0.6 and 1.0. The licensing PCT is obtained as described in Reference S–22.

S.2.2.3.2.5.2 BWR/2

BWR/2s have all been analyzed using SAFER/CORECOOL/GESTR–LOCA on a plant–specific basis. The analysis methodology is described in Reference S–74.

S.2.2.3.2.6 Total LOCA Analysis

The total LOCA analysis, based on the use of the SAFE/REFLOOD/CHASTE codes (Sections S.2.2.3.2.1 and S.2.2.3.2.3), is performed using the procedures outlined in Reference S–19. The total LOCA analysis based on the use of the SAFER/GESTR–LOCA codes (Sections S.2.2.3.2.4 and S.2.2.3.2.5), is performed using the procedures outlined in Reference S–22. The total LOCA analysis is generally provided for each plant independent of the supplemental reload licensing report. The supplemental reload licensing report will contain either the MAPLHGR and PCT as a function of exposure for fuel not previously licensed to operate in the specific reactor, or a reference to the analysis results. For multiple lattice fuel designs, each lattice has an associated MAPLHGR value. The MAPLHGR limit is determined by the LOCA analyses described in the preceding subsections. For each multiple lattice fuel bundle type, the supplemental reload licensing report will include a plot or table of the limiting value of MAPLHGR for the most limiting enriched lattice as a function of average planar exposure. Additional information is provided in Reference S–44.

S.2.2.3.3 Main Steam Line Break Accident Analysis

The analysis of the main steam line break accident depends on the operating thermal–hydraulic parameters of the overall reactor (such as pressure) and overall factors affecting the consequences (such as primary coolant activity). Results for initial cores are documented in the individual plant FSAR. Insertion of the reload fuel designs described in Reference S–2 and S–3 will not change any of these parameters; therefore, the previous reviewed results of this analysis will not change.

S.2.2.3.4 One Recirculation Pump Seizure Accident Analysis

This accident is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft while the reactor is operating at full power.

The pump seizure event is a very mild accident in relation to other accidents such as the LOCA. This is easily verified by consideration of the two events. In both accidents, the recirculation driving loop flow is lost extremely rapidly. In the case of the seizure, stoppage of the pump occurs; for the LOCA, the severance of the line has a similar, but more rapid and severe influence. Following a pump seizure event, flow continues, water level is maintained, the core remains submerged and this provides a continuous core cooling mechanism. However, for the LOCA, complete flow stoppage occurs and the water level decreases due to loss of coolant, resulting in uncovery of the reactor core and subsequent overheating of the fuel rod cladding. In addition, for the pump seizure accident, reactor pressure does not significantly decrease, whereas complete depressurization occurs for the LOCA. Clearly, the increased temperature of the cladding and reduced reactor pressure for the LOCA both combine to yield a much more severe stress and potential for cladding perforation for the LOCA than for the pump seizure. Therefore, it can be concluded that the potential effects of the hypothetical pump seizure accident are very conservatively bounded by the effects of a LOCA and specific analyses of the pump seizure accident are not required.

S.2.2.3.5 Refueling Accident Analysis

Identification of Causes. Accidents that result in the release of radioactive materials directly to the containment can occur when the drywell is open. A survey of the various conditions that could exist when the drywell is open reveals that the greatest potential for the release of radioactive material occurs when the drywell head and reactor vessel head have been removed. In this case, radioactive material released as a result of fuel failure is available for transport directly to the containment.

Various mechanisms for fuel failure under this condition have been investigated. With the current fuel design the refueling interlocks, which impose restrictions on the movement of refueling equipment and control rods, prevent an inadvertent criticality during refueling operations. In addition, the reactor protection system can initiate a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned criticality tests with the reactor vessel head off. It is concluded that the only accident that could result in the release of significant quantities of fission products to the containment during this mode of operation is one resulting from the accidental dropping of a fuel bundle onto the top of the core.

This event occurs under non–operating conditions for the fuel. The key assumption of this postulated occurrence is the inadvertent mechanical damage to the fuel rod cladding as a consequence of the fuel bundle being dropped on the core while in the cold condition. Therefore, fuel densification considerations do not enter into or affect the accident results.

Methods, Assumptions and Conditions. The assumptions and analyses applicable to this type of fuel handling accident are described below.

(1) GE is now manufacturing a new design of the refueling mast with grapple head (NF–500). The new design weighs more—619 pounds compared to 350 pounds. For plants not having employed the new NF–500 refueling mast, the following analysis is bounding.

(2) The number of fuel rods in a fuel bundle has gone from the initial 7x7 array, to the 8x8 array, and more recently to the 9x9 array and the 10x10 array with corresponding dimensional changes.

(3) During a refueling operation a fuel assembly is moved over the top of the core. While the fuel grapple is in the overhoist condition with the bottom of the assembly 34 feet above the top of the core (the maximum height allowed by the fuel handling equipment), a main hoist cable fails allowing the assembly, the fuel grapple mast and head to fall on top of the core impacting a group of four assemblies. The grapple head and mast are fixed vertically to the dropped assembly such that all the kinetic energy is transferred through the dropped assembly to the group of impacted assemblies. The dropped assembly impacts the core at a slight angle and the rods in this assembly are subjected to bending. After the assembly impacts the core, the assembly, grapple head and mast fall onto the core horizontally without contacting the side of the pressure vessel.

(4) The entire amount of potential energy, including the energy of the entire assemblage falling to its side from a vertical position (referenced to the top of the reactor core), is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core and requires that the grapple cable break, allowing the grapple head and three sections of the telescoping mast to remain attached to the falling assembly.

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

(6) All fuel rods, including tie rods, were assumed to fail by 1% strain in compression, the same mode as ordinary fuel rods. For the fuel designs considered here, there is no propensity for preferential failure of tie rods.

The following analysis is provided for the GE11 (Reference S–76) and GE13 fuel bundles (the 9x9 array). The radiological consequences are provided for all fuel designs.

Analysis and Results. Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous prediction of the number of failed rods is not possible. For this reason a simplified energy approach is taken and numerous conservative assumptions are made to assure a conservative estimate of the number of failed rods.

The number of failed fuel rods is determined by balancing the energy of the dropped assemblage against the energy required to fail a rod. The wet weight of the dropped bundle is 562 pounds for the 9x9 fuel rod array bundle (617 pounds for the 7x7 fuel rod array bundle) and wet weight of the grapple mast and head is 619 pounds. The drop distance is 34 feet. The total energy to be dissipated by the first impact is

$$E_1 = (562{+}619)\ (34)\ = 40{,}154\,\mathit{ft\text{–}lb}$$

One half of the energy is considered to be absorbed by the falling assembly and one half by the four impacted assemblies.

No energy is considered to be absorbed by the fuel pellets (i.e., the energy is absorbed entirely by the non–fuel components of the assemblies).

The energy available for clad deformation is considered to be proportional to the mass ratio:

$$\frac{mass\ of\ cladding}{(mass\ of\ assembly) - (mass\ of\ fuel\ pellets)}$$

and is equal to a maximum of 0.510 for the fuel designs considered here.

The energy absorbed by the cladding of the four impacted assemblies is:

$$(20{,}077\,\mathit{ft\text{–}lb})\ (0.510) = 10{,}239\,\mathit{ft\text{–}lb}.$$

Each rod that fails is expected to absorb approximately 200 ft–lb before cladding failure, based on uniform 1% plastic deformation of the cladding.

The number of rods failed in the impacted assemblies is:

$$N_f = \frac{(10{,}239\ ft - lb)}{(200\ ft - lb)} = 51\ rods.$$

The dropped assembly is assumed to impact at a small angle from vertical, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason it is assumed that all the rods in the dropped assembly fail. The total number of failed rods on initial impact is 74 + 51 = 125.

The assembly is assumed to tip over and impact horizontally on the top of the core from a height of one bundle length, approximately 160 inches. The remaining available energy is calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight.

$$E_2 = W_G H_G + \int_0^{H_B} \frac{W_b}{H_B} y dy = W_G H_G + \frac{1}{2} W_B H_B$$

$$E_2 = (619\ lb)\left(\frac{160}{12}\right) + \frac{1}{2}(562)\left(\frac{160}{12}\right) = 12{,}000\ ft - lb.$$

As before, the energy is considered to be absorbed equally by the falling assembly and the impacted assemblies. The fraction available for clad deformation is 0.510. The energy available to deform the unfailed cladding in the impacted assemblies is one–half the energy resulting from the second impact:

$$E_c = (0.5)\ (12{,}000\ ft\text{–}lb)\ (0.510) = 3{,}060\ ft\text{–}lb$$

and the number of failures in the impacted assemblies is:

$$N_F = \frac{3{,}060\ ft - lb}{200\ ft - lb} = 15\ rods.$$

Since the rods in the dropped 9x9 assembly are considered to have failed in the initial impact, the total failed rods resulting from both impacts is 125 + 15 = 140.

The above analysis was completed using the GE12 and GE14 10x10 fuel rod arrays (References S–77 and S–95). The analysis resulted in 172 failed rods from both impacts.

This compares with 111 failed rods from the analysis for the 7x7 fuel rod array bundle presented in the individual plant FSAR.

Radiological Consequences Comparisons. For the purposes of this evaluation, it is conservatively assumed that the fractional plenum activity for any 9x9 rod will be 49/74, or 0.66 times the activity in a 7x7 rod. Based on the assumption that 140 9x9 rods fail compared to 111 for a 7x7 core, the relative amount of activity released for the 9x9 fuel is (140/111) (0.66) = 0.83 times the activity released for a 7x7 core. The activity released to the environment and the radiological exposures for all GE 9x9 fuel designs will therefore be less than 83% of those values presented in the FSAR for a 7x7 core. As identified in the FSAR, the radiological exposures for the 7x7 fuel are well below those guidelines set forth in 10CFR100; therefore, it can be concluded that the consequences of this accident with the new NF–500 mast and the 9x9 fuel will also be well below these guidelines.

A fuel bundle damage analysis and the resulting radiological consequences for the new NF–500 mast and the 8x8 fuel shows that the activity released to the environment and the radiological exposures will be less than 84% of those values presented in the FSAR for a 7x7 core. Similar to the above evaluation, the activity released to the environment and the radiological exposures for all GE 10x10 fuel designs will therefore be less than (172/111)(49/87.33) = 0.87 or 87% of those values presented in the FSAR for a 7x7 core.

S.2.3 Analysis Initial Conditions and Inputs

Inputs to the models utilized to analyze the AOO events discussed in Section S.2.2 are plant unique. The specific inputs related to the plant pressure relief systems (i.e., safety valves, safety/relief valves, etc.) are listed in the supplemental reload licensing report for each plant. Inputs such as thermal power, dome pressure, etc. are given in the individual plant supplemental reload licensing report. The initial conditions for the GETAB analysis are listed in the supplemental reload licensing report for each specific plant. Because the AOO model establishes operating conditions, only licensing basis values are given in the supplemental reload licensing report.

Cycle–dependent initial conditions for the GETAB analysis and the resulting reload parameters are given in the plant FSAR or the supplemental reload licensing report.

S.3 Vessel Pressure ASME Code Compliance Model

The pressure relief system was designed to prevent excessive overpressurization of the primary system process barrier and the pressure vessel and thereby precludes an uncontrolled release of fission products.

Prior to 1967, the design capacities of the safety valves for BWRs were determined according to the requirements of Section I, *Power Boilers*, of the ASME Boiler and Pressure Vessel Code. Under the provisions of this code, safety valve capacities were established to prevent either a vessel or pressure rise greater than 6% above the maximum allowable working pressure. At least one safety valve was to be set at or below the maximum allowable working pressure; the highest safety valve setting could not exceed 103% of the maximum allowable working pressure. No credit was allowed for reactor scram as a complementary pressure protection device. Thus, the required safety valve capacities were sized assuming essentially instantaneous isolation of the pressure vessel with no pressure relief other than that from the safety valves. Nine Mile Point–1 and Oyster Creek are the only plants that were designed to these criteria.

In 1991 Oyster Creek updated its overpressurization analysis (Reference S–88) to ASME Boiler and Pressure Code, Section III to be consistent with later BWRs and reducing the number of safety valves.

In 1995 Nine Mile Point Unit 1 updated its overpressurization analysis (reference S–89) to ASME Boiler and Pressure Code, Section III to be consistent with later BWRs and reduced the number of safety valves.

The vessel overpressure protection system for the other plants was designed to satisfy the requirements of Section III, *Nuclear Vessels*, of the ASME Boiler and Pressure Vessel Code. The ASME Boiler and Pressure Vessel Code, Section III, Class I, permits pressure transients up to 10% over design pressure, and requires that the lowest qualified valve setpoint be at or below the vessel design pressure and the highest setpoint is not greater than 105% of the vessel design pressure. Section III of the code allows credit to be taken for the scram protection system as a pressure protection device when determining the required safety valve capacities for nuclear vessels. As required by the Code of Federal Regulations 10CFR50.55a, paragraph C1, applicable Section III code cases and addenda to which the above plants were designed vary from the 1963 edition, including addenda through summer 1964, to the 1965 edition including addenda through summer 1967. These editions and addenda to Section III of the code required the reactor pressure vessel to be designed to accommodate the normal operating loads and transient startup/shutdown and test cyclic loads expected during the 40–year life of the plant.

In 1968, GE went beyond the code requirements by establishing new design criteria in response to a NRC question. With these criteria, two categories of events (normal and accident) were analyzed for plants that had not received an operating license. The normal category of events included the design and operating loads as well as upset conditions previously analyzed. These loadings were required to meet the criteria documented in Section III of the code. The accident category included low probability of occurrence accidents or faulted conditions that were required to meet a set of limits developed by GE.

The Summer 1968 Addenda to the 1968 Edition of Section III to the ASME code revised the conditions to be considered when performing pressure vessel stress analyses. Loads were to be considered from four categories of conditions: (1) normal; (2) upset; (3) emergency; and (4) faulted.

The Addenda defines an upset condition as any deviation from normal operating conditions caused by any single error, malfunction or a transient which does not result in a forced outage. These events are anticipated to occur frequently enough that design should include the capability to withstand the upsets without operational impairment. Emergency conditions are stated as having "... a low probability of occurrence ..." and require shutdown for correction but cause no gross damage to the system. Additionally, faulted conditions are " ... those combinations of conditions associated with extremely low probability postulated events ..." which may impair the integrity and operability of the nuclear system to the point where public safety is involved.

As documented in later FSARs and accepted by the NRC, GE has defined an upset event as one which has a 40–year encounter probability of occurrence of 10^{-1} through 1; an emergency event has a 40–year encounter probability of 10^{-3} though $<10^{-1}$; and a faulted event has a 40–year encounter probability of 10^{-6} through $<10^{-3}$. GE analyses have determined the probability of occurrence of MSIV closure is 1 event/plant–year (Reference S–48). Failure probability of the direct MSIV position switch scram failure such that scram occurs on neutron monitoring system signal is 1×10^{-3}/demand. Using the above probabilities, this event should be considered an "emergency" condition. Therefore, application of the "emergency" limit under these assumed failure conditions would be considered appropriate. However, in addition to conservatively assuming failure of the direct safety grade position scram signals in its licensing analysis, and conservatively relying upon indirectly derived signals (high neutron flux) from the Reactor Protection System, GE further conservatively applies the upset code requirements, and required pressure safety limits, rather than the more appropriate emergency limits. Application of the direct position scrams in the design basis could be used since they qualify as acceptable pressure protection devices when determining the required safety/relief valve capacity of nuclear vessels under the provisions of the ASME safety code.

As described in the Summer 1968 Addenda of Section III, the following pressure limits are applied to the operating limit category:

(1) Under upset conditions, the code requires that reactor pressures are not to exceed 110% of design pressure ($1.1 \times 1250 = 1375$ psig).

(2) For emergency conditions, it allows up to 120% of design pressure ($1.2 \times 1250 = 1500$ psig).

(3) For faulted conditions, it allows up to 150% of design pressure ($1.5 \times 1250 = 1875$ psig).

GE sensitivity studies (Reference S–49) show the effect of safety/relief valve failures on peak pressure for the MSIV closure event expectedly results in a peak pressure increase of less than 20 psi and depends on the plant total pressure relief capacity.

If an MSIV closure analysis which considers the failure of a safety/relief valve is performed, the following events are considered: (1) MSIV closure followed by indirect flux scram (estimated probability = 1×10^{-3}/demand), and (2) failure of one safety/relief valve. In addition, many conservatisms discussed previously would also be employed. According to the interpretation of the code, MSIV closure with indirect flux scram would be considered an emergency event. Therefore, the occurrence of failures in addition to the extremely low probability of this event constitutes emergency, if not faulted, conditions. Analysis of MSIV closure, flux scram and SRV failure under emergency conditions (1500 psi pressure limit) would be far less restrictive than the present analysis of MSIV closure followed by flux scram under upset conditions (1375 psi pressure limit), especially when considering the minimal effect of a failed SRV.

Overpressurization protection analysis is performed using the ODYN transient code (References S–50 and S–51). In accordance with Reference S–48, no addition of uncertainty to the calculations of pressure is needed. Results for this analysis are given in the FSAR or in the supplemental reload licensing report.

S.4 Stability Analysis Methods

Two types of stability analyses are performed generically to ensure continued acceptable plant–specific implementation of NRC approved long–term stability solutions:

- Core and channel decay ratio calculations are performed to ensure that the fuel is as stable as previously licensed GE fuel designs. If the fuel is not as stable as previously existing fuel designs, then the stability exclusion region must be revised to provide the same level of protection.

- CPR response calculations are performed to demonstrate that the generic DIVOM curve (**D**elta CPR over **I**nitial CPR **V**s. **O**scillation **M**agnitude) defined in Reference S–85 is applicable. If the generic DIVOM curve is not applicable, then a new fuel specific DIVOM curve must be defined.

The core and channel decay ratios are calculated with a NRC approved frequency domain model. This calculation provides assurance that plants with prevention based long–term stability solutions will not have to unreasonably increase the size of their stability–based regions for the evaluated fuel design.

The continued applicability of the interim/backup stability solution is based on exclusion regions and reload validation of these exclusion regions as required to ensure full stability protection.

The applicability of the generic DIVOM curve is demonstrated with a best–estimate coupled neutronic – thermal hydraulic model. This is the same model that was used to generate the generic DIVOM curve. The DIVOM curve is required for plants with a detect and suppress solution to demonstrate safety limit MCPR compliance.

S.4.1 BWROG Long–Term Stability Solutions

S.4.1.1 Enhanced Option I–A

Enhanced Option I–A (EIA) is a prevention solution. EIA was reviewed and approved by the USNRC as documented in References S–80 through S–84 and Reference S–96. For plants implementing EIA, the prescribed reload validation (Reference S–80) is performed each cycle and the results documented in the supplemental reload licensing report. The validation confirms that the existing E1A stability regions provide adequate stability margin. If E1A reload validation criteria are not met, new E1A stability regions must be defined and implemented.

S.4.1.2 Option II

Option II is a combination prevention and detect and suppress solution. Option II was reviewed and approved by the USNRC as documented in Reference S–79. Option II is only applicable to BWR 2 plants. A reload review criterion has been defined for Option II to ensure that the existing exclusion region is acceptable for each fuel cycle. If reload criteria are not met, the exclusion region must be recalculated. In addition, continued safety limit MCPR protection is demonstrated for each fuel cycle using the methodology documented in Reference S–85 and in plant–specific Option II licensing topical reports. The results of the reload review and safety limit MCPR protection calculation are documented in the supplemental reload licensing report.

S.4.1.3 Option I–D

Option I–D is a combination prevention and detect and suppress solution. Option I–D was reviewed and approved by the USNRC as documented in Reference S–79. Option I–D is only applicable to plants which can demonstrate that the core wide is the predominate oscillation mode for anticipated reactor instabilities. A reload review criterion has been defined for Option I–D to ensure that the existing exclusion region is acceptable and that the safety limit MCPR is protected for each fuel cycle. If reload criteria are not met, the exclusion region must be recalculated. In addition, continued safety limit MCPR protection is demonstrated for each fuel cycle using the methodology documented in Reference S–85. The dominance of the core–wide mode of reactor oscillation is demonstrated at the most limiting power/flow point using the NRC–approved frequency stability code (e.g., Reference S–96). The results of the reload review and safety limit MCPR protection calculation are documented in the supplemental reload licensing report.

S.4.1.4 Option III

Option III is a detect and suppress solution. Option III was reviewed and approved by the USNRC as documented in Reference S–79. Continued safety limit MCPR protection is demonstrated for each fuel cycle using the methodology documented in Reference S–85. The results of the safety limit MCPR protection calculation are documented in the supplemental reload licensing report.

S.4.2 Interim/Backup Stability Solution

S.4.2.1 Interim Corrective Action (ICA)

The ICA is an interim prevention solution based on exclusion regions. The currently used ICA regions were established in Reference S–91 based on original licensed thermal power, generally shorter fuel cycles, and more stable core designs. These regions are defined based on relative core flow and rod line points and not on specific stability criteria. New aggressive core design changes may have reduced stability margins. GE recommends that the impact of core design changes be included in plant/cycle–specific evaluations to assess the continued applicability of the ICA regions. The results of the ICA analysis are documented in the supplemental reload licensing report.

S.5 Analysis Options

Three groups of analysis options are presented in the following sections. The first group involves options that may be chosen to improve MCPR margin. The second group of improvements represents a collection of possible operating flexibility options. **Also noted in the second group is the GE Licensing Topical Report, *Applicability of GE Methods to Expanded Operating Domains* (Reference S-102), which may be part of the licensing basis for EPU and MELLLA+ plants.** The third group includes the requirements for applying the generic analysis in Reference S-99 for the Fuel Loading Error event. In some cases separate plant specific reports are submitted for approval before the option is available. Other options are supported by generic analyses that have been approved and only require that the plant choose to activate the option. In each case, the plant options are selected for each cycle and documented in the cycle design documentation and the plant supplemental reload licensing report (SRLR).

S.5.1 Available MCPR Margin Improvement Options

The following margin improvement options have been developed for operating BWRs:

(1) Recirculation Pump Trip

(2) Rod Withdrawal Limiter

(3) Thermal Power Monitor

(4) Exposure–Dependent Limits

(5) Improved Scram Times
   (a) Measured Scram Time
   (b) Generic Statistical Scram Time (ODYN Option B or TRACG Option B)

These margin improvement options will be made available, on a commercial basis, to all owners of operating BWRs.

As these options are selected by the BWR owners, plant–specific and/or generic bounding analyses will be submitted for approval. The plant supplemental reload licensing report will designate the options selected by that BWR owner.

S.5.1.1 Recirculation Pump Trip

For many of the plant operating cycles, the limiting AOOs are the turbine trip, generator load rejection, or other AOOs that result in a turbine trip. A significant improvement in thermal margin can be realized if the severity of these transients is reduced. The Recirculation Pump Trip (RPT) feature accomplishes this by cutting off power to the recirculation pump motors anytime that the turbine control valve or turbine stop valve fast closure occurs. This rapid reduction in recirculation flow increases the core void content during the AOO, thereby reducing the peak AOO power and heat flux.

Basically, the RPT consists of switches installed in both the turbine control valves and the turbine stop valves. When these valves close, breakers are tripped between the MG sets and the recirculation pump motors; this releases the recirculation pumps to coast down under their own inertia.

Recirculation pump trip is standard equipment in all later plants.

S.5.1.2 Rod Withdrawal Limiter System

The Rod Withdrawal Error (RWE) has become the limiting transient for some plants. A new Rod Withdrawal Limiter System (RWLS) concept has been developed. This new system will restrict control rod movement such that the Rod Withdrawal Error will be eliminated as a limiting AOO.

The RWLS functions by providing a rod withdrawal block as a function of rod distance traveled per rod selection. Core physics calculations performed for the RWE analysis, provide the decrease in CPR as a function of rod travel. After choosing an acceptable ΔCPR, an allowable rod movement is determined. This sets the RWLS trip point. Any attempt to withdraw the rod by more than the trip point results in a rod block. Thus, an upper bound is established on the CPR decrease that can result from any single rod withdrawal error.

This system is standard on all BWR/6 plants.

S.5.1.3 Thermal Power Monitor

The APRM simulated thermal power trip (APRM thermal power monitor) is a minor modification to the APRM system. The modified APRM system generates two upscale trips. On one, the APRM signal (which is proportional to the thermal neutron flux) is compared with a reference that is not dependent on flow rate.

During normal reactor operations, neutron flux spikes may occur due to conditions such as transients in the recirculation system, transients during large flow control load maneuvers, transients during turbine stop valve tests and transients in plants with equalizer lines when the recirculation equalizer lines are opened. The neutron flux leads the heat flux during transients because of the fuel time constant. And the neutron flux for these transients trips upscale before the heat flux increases significantly. (High heat flux is the precursor of fuel damage.) Thus, increased availability can be achieved without fuel jeopardy by adding a trip dependent on heat flux (thermal power).

For this trip, the APRM signal is passed through a low pass RC filter. It is compared with a recirculation flow rate dependent reference that decreases approximately parallel to the flow control lines.

In addition to increased availability, another benefit is that with the minor operational spikes filtered out, the heat flux trip setpoint is lower than the neutron flux trip setpoint. For long, slow AOOs such as the loss–of–feedwater heater, the heat flux and neutron flux are almost in equilibrium. For these AOOs, the lower trip setpoint results in an earlier scram with a smaller increase in heat flux and a corresponding reduction in the consequences.

The APRM Simulated Thermal Power Trip is standard equipment in all current pre–operational plants.

S.5.1.4 Exposure–Dependent Limits

The severity of any plant AOO pressurization event is worst at the end of the cycle primarily because the EOC all–rods–out scram curve gives the worst possible scram response. It follows that some limits relief may be obtained by analyzing the AOOs at other interim points in the cycle and administering the resulting limits on an “exposure dependent” basis.

This technique is straightforward and consists merely of repeating certain elements of the AOO analyses for selected mid–cycle exposures. Because the scram reactivity function monotonically deteriorates with exposure (after the reactivity peak), the limit determined for an exposure E_1 is administered for all exposures in the interval $E_{i-1} < E \leq E_i$ where E_{i-1} is the next lower exposure point for which a limit was determined. This results in a table of MCPR limits to be applied through different exposure intervals of the cycle.

S.5.1.5 Improved Scram Times

S.5.1.5.1 Measured Scram Time

Control rod scram time data from two operating BWR/4 plants have been used to derive a more realistic scram insertion time specification to be used in plant AOO analyses. The total database exceeds 1600 rod scram times. The primary impact of measured scram time is in the plant pressure/power increase AOOs and feedwater controller failure. To use this option, a plant must show that the actual plant control rod insertion time (plus three standard deviations) is within the above more realistic specification or another derived scram time specification. Operating limits for plants taking credit for measured scram time are determined using either GENESIS, GEMINI or TRACG methods and procedures.

S.5.1.5.2 Generic Statistical Scram Time (ODYN Option B or TRACG Option B)

GE has developed a generic statistical scram time distribution for the purposes of generating the AOO ΔCPR adjustment factors required for ODYN Option B or TRACG Option B (see Section 4.0 of Reference S–1). Those plants operating under Option B MCPR operating limits will be taking advantage of the improved scram time benefits on the AOO performance, by demonstrating that actual scram speeds conform with the generic statistical scram times assumed. Operating limits for plants taking credit for the generic statistical scram time are determined using either GENESIS, GEMINI or TRACG methods and procedures.

S.5.2 Operating Flexibility Options

The following operating flexibility options have been developed for BWRs:

(1) Single–Loop Operation.

(2) Load Line Limit.

(3) Extended Load Line Limit.

(4) Increased Core Flow.

(5) Feedwater Temperature Reduction.

(6) ARTS Program (BWR/3–5).

(7) Maximum Extended Operating Domain for BWR/6 and Maximum Extended Load Line Limit Analysis for BWR/3–5.

(8) Turbine Bypass Out of Service.

(9) Safety/Relief Valves Out of Service.

(10) ADS Valve Out of Service.

(11) End–of–Cycle Recirculation Pump Trip Out of Service.

(12) Main Steam Isolation Valves Out of Service.

The supplemental reload licensing report indicates if an option has been chosen.

Some plants referencing GESTAR II as the applied reload methodology may include the GE Licensing Topical Report, Applicability of GE Methods to Expanded Operating Domains (Reference S-102), as part of their licensing basis. For such a plant, the limitations, conditions, and requirements of Reference S-102 are included in the analysis and licensing basis for the reload.

S.5.2.1 Single–Loop Operation

Technical Specifications for a plant without a Single–Loop Operation (SLO) analysis do not allow operation beyond a relatively short period of time if an idle recirculation loop cannot be returned to service. Typically, the plant shall not be operated for a period in excess of 24 hours with one recirculation loop out of service.

The capability of operating at reduced power with a single recirculation loop is highly desirable, from a plant availability/outage planning standpoint, in the event maintenance of a recirculation pump or other components renders one loop inoperative. The SLO analysis evaluates the plant for continuous operation at a maximum expected power output that is 20% to 30% below that which is attainable for two–pump operation.

To justify SLO, safety analyses have to be reviewed for one–pump operation. The MCPR fuel cladding integrity safety limit, AOO analyses, operating limit MCPR, and non–LOCA accidents are evaluated. Increased uncertainties in the total core flow and traversing in–core probe (TIP) readings result in a small increase in the fuel cladding integrity safety limit MCPR.

SLO can also result in changes to plant response during a LOCA. These changes are accommodated by the application of reduction factors to the two–loop operation MAPLHGRs if required. MAPLHGR reduction factors are evaluated on a plant–by–plant and fuel type dependent basis. In each subsequent reload, reduction factors are checked for validity and, if new fuel types are added, new reduction factors may be needed in order to maintain the validity of the SLO analysis.

S.5.2.2 Load Line Limit (BWR/2–4 Only)

For non–barrier fuel, fuel pellet–cladding interaction considerations inhibit withdrawal of control rods at high power levels. In order to attain rated power and not exceed rated core flow without control rod withdrawals at high power when using non–barrier fuel, operation above the rated load line is required during power ascension. Consequently, an analysis referred to as the Load Line Limit Analysis (LLLA) is performed to determine if the safety consequences of operation above the rated load line, but within a defined region of the power flow map, are bounded by the respective consequences of operation at the licensing basis conditions.

The region above the rated load line is known as the extended operating region and is defined by the locus of power/flow points bounded by:

(1) the rated load line;

(2) the APRM rod block line; and

(3) the rod block intercept line (BWR/2 and 3), or the rod block intercept line and the rated power line (BWR/4).

LLLA is performed on a plant/cycle–specific basis. However, after the LLLA is initially performed for a plant and cycle, on subsequent cycles only the following checks need to be made in addition to the standard reload analyses to support operation in the extended operation region:

(1) **LOCA** – The applicability of previous LOCA analyses to the extended operating region must be verified for each plant during each cycle.

(2) **AOOs** – The consequences of AOOs are evaluated to determine if operating limit adjustments are necessary for operation in the extended operating range.

BWR/5 and 6 are designed with expanded operating flexibility that supports plant operation in an extended region above the rated load line up to rated power. This expanded flexibility is validated whenever a fuel design with different transient response characteristics is introduced.

S.5.2.3 Extended Load Line Limit (BWR/2–6)

The Extended Load Line Limit Analysis (ELLLA) is similar to the LLLA described in Subsection S.5.2.2. However, the extended operating domain for ELLLA, instead, has an upper bound of the APRM rod block line to rated power for BWR/2–6.

Once ELLLA has been performed for a specific plant and cycle, it is reverified for applicability to subsequent cycles as described in Sub–section S.5.2.2. Because of the different extended operating regions for ELLLA and LLLA, the power/flow points chosen for analysis may be different.

Some plants have, in plant specific submittals, relaxed the APRM rod block setpoints. For these plants, the ELLLA region no longer corresponds to the APRM rod block line. The APRM setpoints and the analyzed operating domain are defined in the plant specific licensing documentation.

S.5.2.4 Increased Core Flow (ICF) Operation

Analyses are performed in order to justify operation at core flow rates in excess of the 100% rated flow condition. The analyses are done for application through the cycle or for application at the end of cycle only.

The limiting AOOs that are analyzed at rated flow as part of the supplemental reload licensing report are reanalyzed for increased core flow operation. In addition, the loss–of–coolant accident (LOCA), fuel loading error evaluated as an AOO only, rod drop accident, and rod withdrawal error are also re–evaluated for increased flow operation to assure that the higher flow and exposure capability does not significantly impact these analyses.

The effects of the increased pressure differences on the reactor internal components, fuel channels, and fuel bundles as a result of the increased flow are analyzed in order to ensure that the design limits will not be exceeded.

The thermal–hydraulic stability is re–evaluated for increased core flow operation, and the effects of flow–induced vibration are also evaluated to assure that the vibration criteria will not be exceeded.

S.5.2.5 Feedwater Temperature Reduction (FWTR)

Analyses are performed in order to justify operation at a reduced feedwater temperature at rated thermal power. Usually, the analyses are performed for end–of–cycle operation with the last–stage feedwater heaters valved out in order to increase the core rated power exposure capability. However, throughout cycle operation, some feedwater temperature reduction can be justified by analyses at the appropriate operating conditions for accommodating the potential of a feedwater heater being out of service.

The limiting AOOs are reanalyzed for operation at a reduced feedwater temperature. In addition, the loss–of–coolant accident (LOCA), fuel loading error evaluated as an AOO only, rod drop accident, and rod withdrawal error are also re–evaluated for operation at a reduced feedwater temperature to assure that the higher subcooling and exposure capability does not significantly impact these analyses.

The reactor core and thermal–hydraulic stability are re–evaluated, along with the increase in the feedwater nozzle fatigue usage factor, for operation at a reduced feedwater temperature throughout the cycle.

S.5.2.6 ARTS Program (BWR/3–5)

The ARTS program is a comprehensive project involving the Average Power Range Monitor (APRM), the Rod Block Monitor (RBM), and Technical Specification improvements.

Implementing the ARTS program provides for the following improvements that enhance the flexibility of the BWR during power level monitoring.

(1) The average power range monitor (APRM) trip setdown requirement is replaced by a power–dependent MCPR operating limit similar to that used in the BWR/6, and a flow–dependent MCPR operating limit to reduce the need for manual setpoint adjustments. In addition, another set of LHGR power– and flow–dependent limits will also be specified for more rigorous fuel thermal protection during postulated transients at off–rated conditions. These power– and flow–dependent limits are

verified for plant–specific application during the initial ARTS licensing implementation and are applicable to subsequent cycles provided that there are no changes to the plant configuration as assumed in the licensing analyses. A plant may also include the power– and flow–dependent limits for MAPLHGR.

(2) The RBM system may be modified from flow–biased to power–dependent trips to allow the use of a new generic non–limiting analysis for the rod withdrawal error (RWE) and to improve response predictability to reduce the frequency of nonessential alarms. The applicability of the generic RWE analysis to GE fuel designs is discussed in Reference S–2.

The resulting improvements in the flexibility of the BWR provided by ARTS are designed to significantly minimize the time to achieve full power from startup conditions.

S.5.2.7 Maximum Extended Operating Domain for BWR/6 and Maximum Extended Load Line Limit Analysis for BWR/3–5

The modified operating envelope termed Maximum Extended Operating Domain (MEOD) for BWR/6 plants permits extension of operation into higher load line power/flow areas, provides improved power ascension capability to full power and additional flow range at rated power, and includes an increased flow region to compensate for reactivity reduction due to exposure during an operating cycle. Overall, MEOD can be utilized to increase operating flexibility and plant capacity factor. The higher load line aspect of MEOD is also applied to BWR/3–5 plants as a Maximum Extended Load Line Limit Analysis (MELLLA). The higher core flow aspect of MEOD is also applied to BWR/3–5 plants as an Increased Core Flow (ICF) Analysis (see Section S.5.2.4).

The extended load line region boundary of MEOD is typically limited to 75% core flow at 100% of the original plant licensed thermal power and the corresponding power/flow constant rod line. The increased–core–flow region is defined on a plant-specific basis (typically between 105 and 110% of rated core flow) and is limited by plant recirculation system capability, acceptable flow–induced vibration, fuel lift considerations, and force impact on the vessel internal components.

Evaluations performed for MEOD conditions include normal and AOOs, LOCA analysis, containment responses, stability, flow–induced vibration, and the effects of increased flow–induced loads on reactor internal components and fuel channels. The limiting AOOs applicable to each plant basis are evaluated for the normal range of operating power and flow conditions. The AOO analyses results are used to establish power and flow dependent MAPLHGR (or LHGR) limits to replace the APRM trip setdown requirement for protection at off–rated power and flow conditions. Also, the power and flow dependent MCPR limits are revised to incorporate the results of the AOO analyses. The MEOD power and flow dependent limits are evaluated for application to follow–on cycles.

S.5.2.8 Turbine Bypass Out of Service (TBOOS)

Some plant technical specifications require surveillance testing of the turbine bypass system response time. Operation of the turbine bypass system is assumed in the analysis of the feedwater controller failure–maximum demand event (see Section S.2.2.1.6). If this event is limiting or near limiting, the operating limit MCPR basis may be invalid if the bypass system cannot be demonstrated to meet response time requirements. Reload evaluations may incorporate a FWCF without credit for bypass operation calculation as a provision when required bypass surveillance cannot be performed, or other temporary factors render the system unavailable. Additionally, for extended operation with degraded bypass system operation, evaluations in support of this condition are augmented with the appropriate limiting events, such as the FWCF, for the applicable cycle.

S.5.2.9 Safety/Relief Valves Out of Service.

This option provides support to operate the plant with one or more safety and/or relief valves declared out of service and is normally included with the SRV setpoint tolerance increase (References S–97 and S–98). The analysis shall include the vessel overpressure, fuel thermal limits, fuel performance during ECCS–LOCA events, high pressure systems performance (HPCS, RCIC, SLCS) and responses to Anticipated Transients Without Scram.

S.5.2.10 ADS Valve Out of Service.

This option provides justification for continuous operation with the automatic depressurization function of one automatic depressurization valve declared out of service. This contingency analysis shall allow flexibility when complying with the technical specification for continuous operation at full power with one ADS valve declared out of service.

S.5.2.11 End–of–Cycle Recirculation Pump Trip Out of Service.

In the event that the end–of–cycle recirculation pump trip becomes inoperable and is therefore not capable of performing its intended function (a recirculation pump trip during specific AOOs), operation can continue at full power when this option is included. Specific AOOs that are terminated by scram due to turbine control valve or turbine stop valve closure will be analyzed without credit to having the recirculation pumps trip system operable.

S.5.2.12 Main Steam Isolation Valves Out of Service.

This option provides justification for continuous operation with a main steam isolation valve out of service when there is not compliance with the requirements of the technical specifications for the main steam isolation valves closure characteristics. The analyses include: fuel thermal limits analysis, vessel overpressure, fuel performance during events of ECCS–LOCA, and analysis of operational aspects, such as margin or adjustment to main steam high flow.

S.5.3 Fuel Loading Error Analysis Requirements

Since 1978, the fuel loading error (FLE) has been analyzed as an AOO and, as such, the change in CPR for the event has been factored into the determination of the MCPR operating limit for each cycle. Section 6.3 of the GESTAR Rev 0 SER May 12, 1978 (Appendix C, Pg. US.C-4) describes the basis for this treatment of the FLE, which includes fuel-loading experience in that time period. In 1981, utilities began improving the procedures used for core verification following refueling. As shown in Reference S-99, the fuel loading error rate for the recent 25-year period and the trend for the most recent 10 years of refueling outages support the classification of the FLE event as an "Infrequent Incident." Section S.2.1 provides the basis for categorizing the FLE as an Infrequent Incident and the analysis limits.

The FLE will be analyzed as an Infrequent Incident provided that the plant confirms the requirements for application of the generic analysis. Should the plant be unable to confirm the requirements, the FLE will be evaluated to meet the fuel cladding integrity safety limit MCPR. Several items must be confirmed and documented through the reload design documentation. The first confirmation involves the core verification procedures applied following refueling, and the second involves the basis for the dose analyses and plant off-gas system bases used to perform the generic radiological analysis. The requirements apply for plants with either 10CFR100 or 10CFR50.67 radiological licensing bases.

Core Verification

The application of the Reference S-99 basis for the FLE requires that plant's core verification procedures must be consistent with those generally used during the recent historical period forming the basis for the Amendment 28 analysis of the event frequency. Therefore, the plant must confirm that their core verification procedures have the following characteristics:

1. During fuel movement, each move (location, orientation, and seating) is observed and checked at the time of completion by the operator and a spotter.
2. After completion of the core load, the core is verified by video recording the core using an underwater camera. The recording may involve two or more records made at different ranges to: provide clear resolution of the bundle serial number, illustrate the orientation in four bundle clusters, and illustrate the proper seating of the bundles. The core verification may take place during the recording process, by viewing after recording, or a combination.
3. Two independent reviewers perform the verification of the bundle serial number (location) and orientation. Each independent review records the bundle serial numbers on a core map, which is verified with the planned as loaded core.

Offsite Radiological Analysis

The plant Chi/Q values used in the applicability confirmation should represent limiting design basis accident Chi/Q values calculated using NRC guidance such as Regulatory Guide (RG) 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, or other methods specifically approved by the NRC for offsite dose analysis at the plant site. The offsite radiological analysis depends on the plant configuration:

Scenario 1 - Plants that have a main steam line high radiation isolation trip.

For plants with a 10CFR100 radiological basis, the limiting 2-hour Chi/Q value at the exclusion area boundary (EAB) is 1.67×10^{-3} s/m^3. Therefore, the plant must confirm that the 2-hour Chi/Q value at the EAB is less than or equal to this value. Any dispersion coefficient less than this value will result in a dose less than the Thyroid 30 Rem limit.

For plants with a 10CFR50.67 radiological basis, the limiting 2-hour Chi/Q value at the exclusion area boundary (EAB) is 5.04×10^{-3} s/m^3. Therefore, the plant must confirm that the 2-hour Chi/Q value at the EAB is less than or equal to this value. Any dispersion coefficient less than this value will result in a dose less than the TEDE 2.5 Rem limit.

Scenario 2 - Plants that do not have a main steam line high radiation isolation trip.

Scenario 2 requires that the plant have an augmented offgas system with the capability to remove iodine indefinitely. The design capability of the augmented offgas system must be confirmed by Scenario 2 plants.

For plants with a 10CFR100 radiological basis, Figures S-3 and S-4 will be used to confirm the applicability of the generic analysis. Three parameters are needed to use these figures: the 2-hour Chi/Q value at the EAB and the hold-up time for krypton and xenon.

The following is an example of determining the dose to be compared to the limit:

Low temperature offgas systems supplied by GE provide minimum decay times of 46 hours for krypton and 42 days for xenon, at the design basis air in-leakage rate of 30 cubic feet per minute. For these decay times, the doses from Figures S-3 and S-4 for the 2-hour Chi/Q at the EAB value of 3×10^{-4} are approximately 1.6×10^{-3} and 7.9×10^{-3} for the krypton and xenon, respectively. Summing these results in an approximate total of 9.5×10^{-3} Rem, which is much less than the 2.5 Rem whole body dose limit. Using the plant specific parameters, the plant must confirm that the plant specific result is less than the 2.5 Rem whole body dose limit.

In a similar fashion, plants with a 10CFR50.67 radiological basis will use Figures S-5 and S-6 to confirm the applicability of the generic analysis. Using the plant specific parameters, the plant must confirm that the plant specific result is less than the 2.5 Rem TEDE dose limit.

Control Room Radiological Analysis

The control room Chi/Q values reported for use in the applicability confirmation should represent limiting design basis accident Chi/Q values calculated using NRC guidance such as Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," or other methods specifically approved by the NRC for control room dose analysis at the plant site.

For plants with a 10CFR100 radiological basis, the maximum allowable control room Chi/Q value is 1.81×10^{-3} s/m^3. Therefore, the plant must confirm that the maximum control room

Chi/Q value is less than or equal to this value. Any dispersion coefficient less than this value will result in a dose less than the 30 Rem Thyroid limit.

For plants with a 10CFR50.67 radiological basis, the maximum allowable control room Chi/Q value is 1.25×10^{-2} s/m^3. Therefore, the plant must confirm that the maximum control room Chi/Q value is less than or equal to this value. Any dispersion coefficient less than this value will result in a dose less than the 5.0 Rem TEDE limit.

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Table S–1

Sensitivity of CPR to Various Thermal–Hydraulic Parameters

Parameter	Approximate Nominal Value	ΔCPR/Nominal CPR (ΔParameter/Nominal Parameter)
Bundle Power (or Relative Bundle Power)	6–6.7 MWt	0 to –1.0
Bundle Coolant Flow	G = 1.1 x 10^6 lbm/hr–ft^2	+0.2 (BWR/4)
Core Coolant Inlet Subcooling	20–27 Btu/lbm	+0.1
R–factor	1.04–1.10	–2.1
Core Pressure (with constant coolant subcooling)	1,035–1,055 psia	–0.6

Table S–2

Plants for which ATWS Pump Trip is Assumed in Transient Analyses

Duane Arnold	Cooper	Fitzpatrick	Hatch 1 & 2
Brunswick 1 & 2	Peach Bottom 2 & 3	Browns Ferry 1, 2, & 3	Vermont Yankee
Pilgrim	Millstone	Dresden 2 & 3	Quad Cities 1 & 2
Monticello	Fermi 2	Hope Creek 1 & 2	Limerick 1 & 2
Shoreham	Susquehanna 1 & 2	Hanford 2	LaSalle 1 & 2
Nine Mile Point 1 & 2	Clinton 1	Grand Gulf 1 & 2	Perry 1 & 2
River Bend 1	Oyster Creek		

Table S–3

ΔCPR as a Function of RBM Setpoint for Generic Rod Withdrawal Error Analysis

RBM Setpoint	ΔCPR
104	0.13
105	0.16
106	0.19
107	0.22
108	0.28
109	0.32
110	0.36

Table S–4

Group Notch Plants[2]

Browns Ferry 1, 2, & 3	Peach Bottom 2 & 3
Fitzpatrick	Cooper
Duane Arnold	Hatch 1 & 2
Brunswick 1 & 2	Fermi 2

[2] Plants that have implemented the requirements described in Reference S–9 or S–10 are no longer classified as Group Notch plants.

Table S–5

Specific Plant Analysis

Plant	Analysis Basis	Specific Plant LOCA Analysis Document	Reference Lead Plant LOCA Analysis Document
Nine Mile Point 1	SAFER/GESTR–LOCA	S–53	N/A
Nine Mile Point 2	SAFER/GESTR–LOCA	S–72	N/A
Dresden 2 and 3	SAFER/GESTR–LOCA	S–54	N/A
Quad Cities 1 and 2	SAFER/GESTR–LOCA	S–54	N/A
LaSalle 1 and 2	SAFER/GESTR–LOCA	S–73	N/A
Monticello	SAFER/GESTR–LOCA	S–55	N/A
Fermi 2	SAFER/GESTR–LOCA	S–56	N/A
Duane Arnold	SAFER/GESTR–LOCA	S–57	N/A
Pilgrim	SAFER/GESTR–LOCA	S–58	N/A
Browns Ferry 1, 2 and 3	SAFER/GESTR–LOCA	S–59	N/A
Hope Creek	**SAFER/GESTR–LOCA**~~SAFE/REFLOOD~~	S–60	N/A
Fitzpatrick	SAFER/GESTR–LOCA	S–71	N/A
Cooper	SAFER/GESTR–LOCA	S–61	N/A
Hatch 1 and 2	SAFER/GESTR–LOCA	S–62	N/A
Brunswick 1 and 2	SAFER/GESTR–LOCA	S–63	N/A
Clinton	SAFER/GESTR–LOCA	S–64	N/A
Vermont Yankee	SAFER/GESTR–LOCA	S–65	N/A
River Bend	SAFER/GESTR–LOCA	S–66	N/A
Limerick 1 and 2	SAFER/GESTR–LOCA	S–67	N/A
Peach Bottom 2 and 3	SAFER/GESTR–LOCA	S–68	N/A
Perry	SAFER/GESTR–LOCA	S–69	N/A
Oyster Creek	SAFER/GESTR–LOCA	S–70	N/A
Susquehanna 1 and 2	SAFER/GESTR–LOCA	S–75	N/A
WNP–2	SAFER/GESTR–LOCA	S–86	N/A
Grand Gulf	SAFER/GESTR–LOCA	S–87	N/A

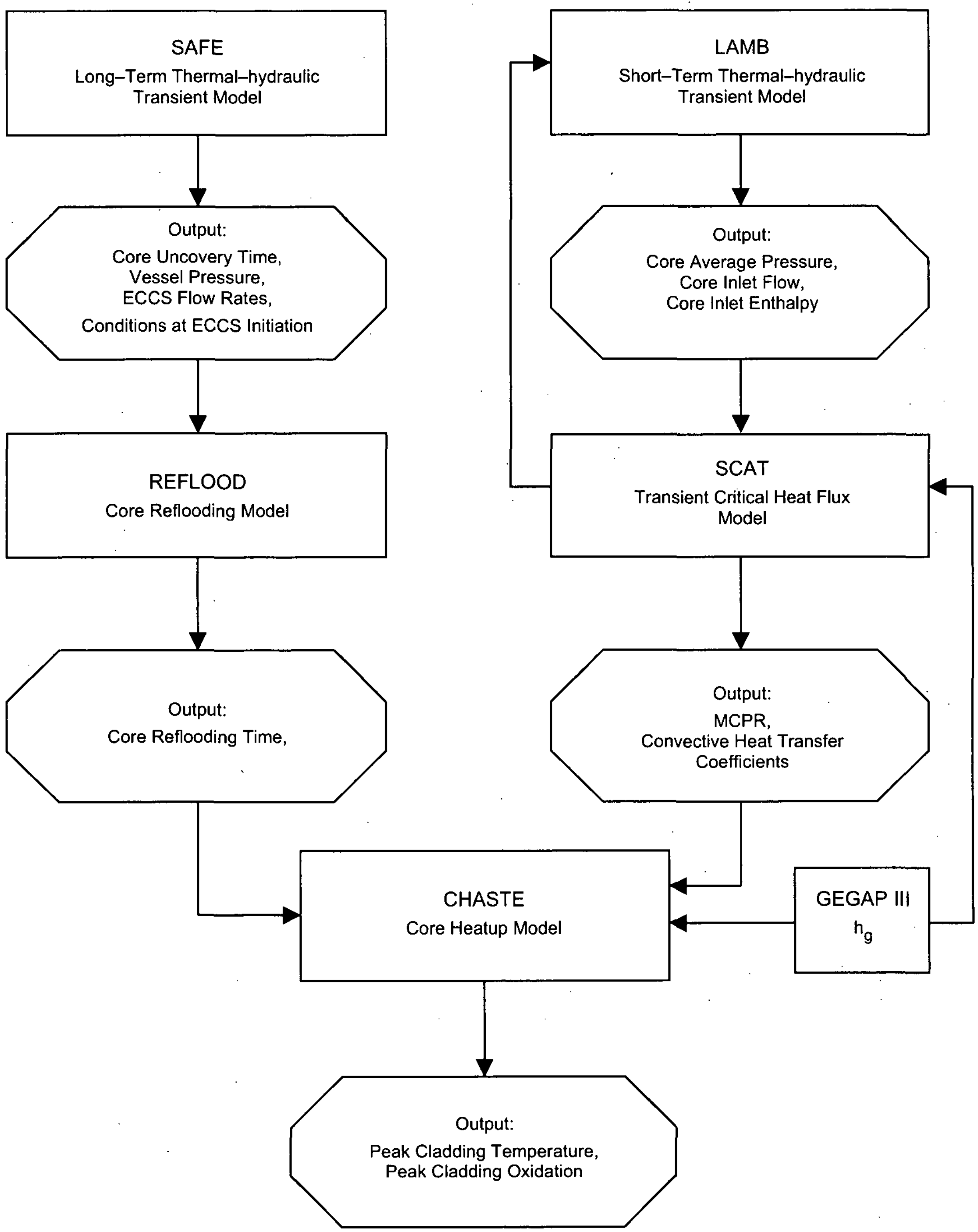

Figure S–1. Loss–of–Coolant Accident Evaluation Model (SAFE/REFLOOD Analysis Methods

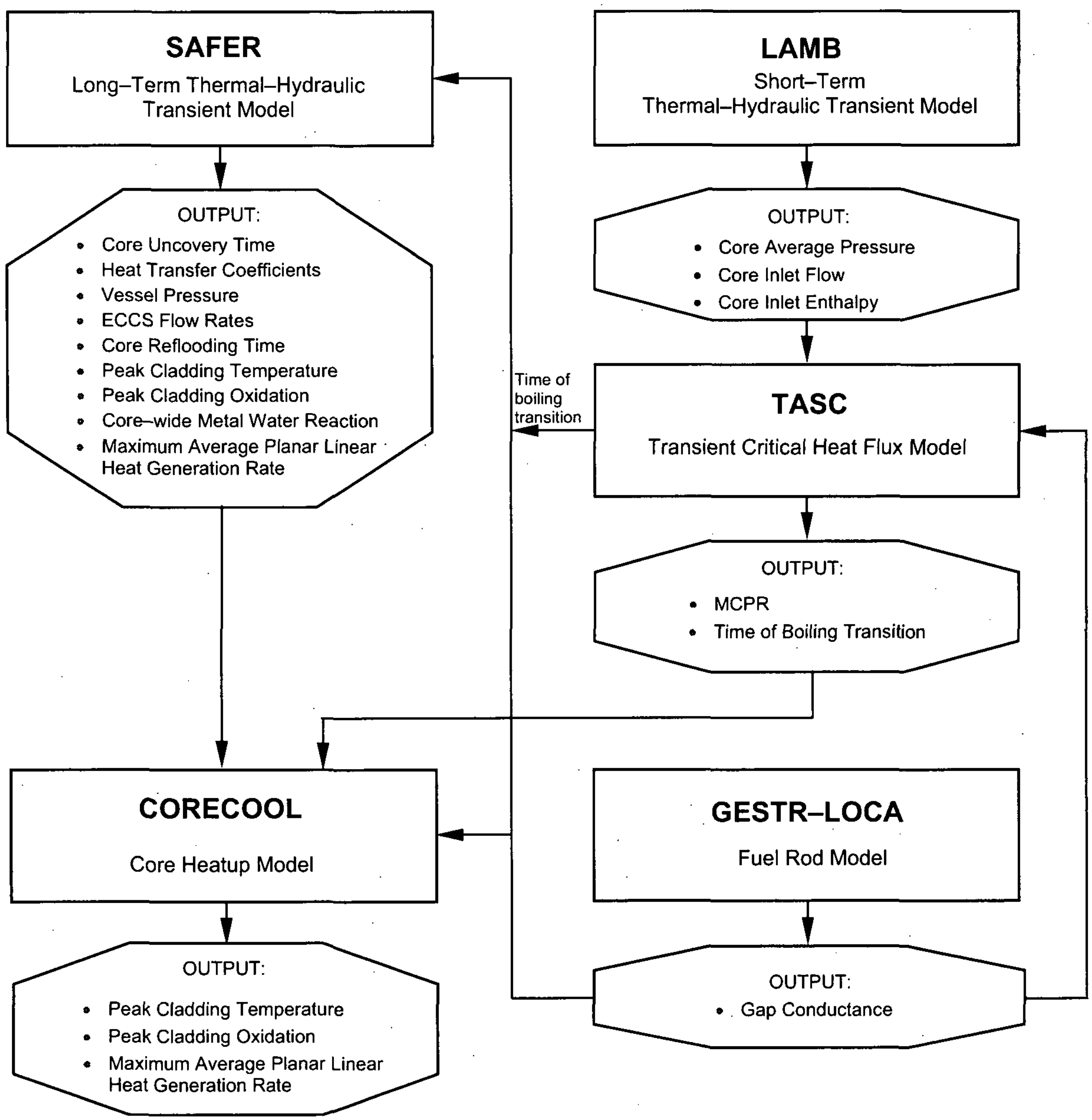

Figure S–2. Loss–of–Coolant Accident Evaluation Model (SAFER/GESTR Analysis Methods)

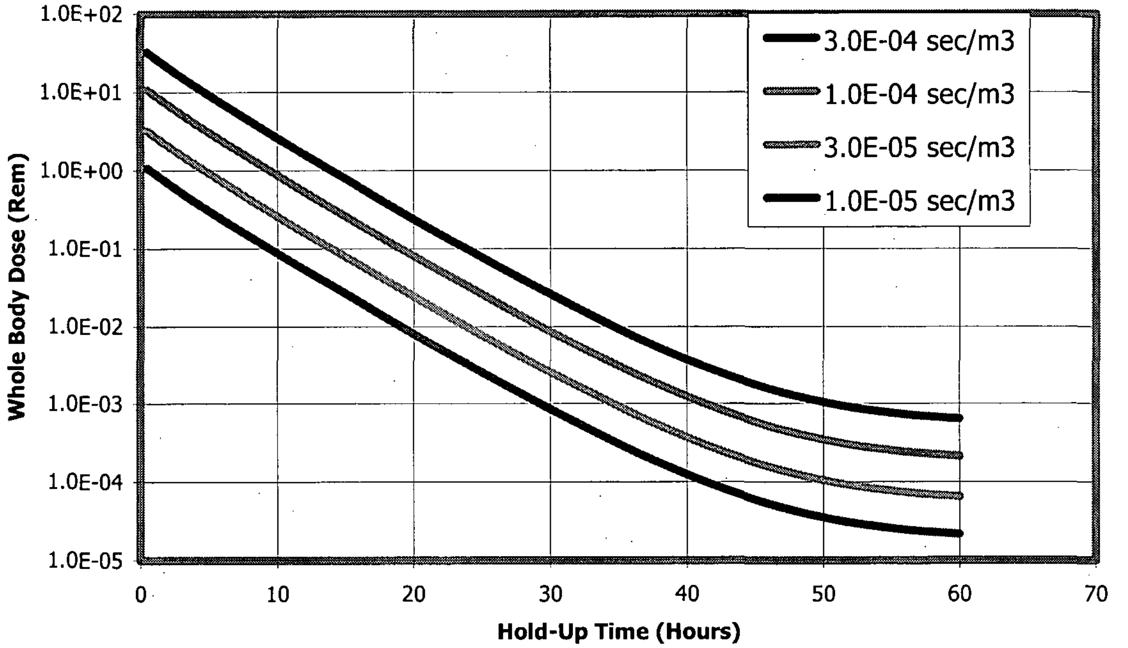

Figure S-3 Scenario 2 Krypton Whole Body Dose with Respect to Charcoal Hold Up

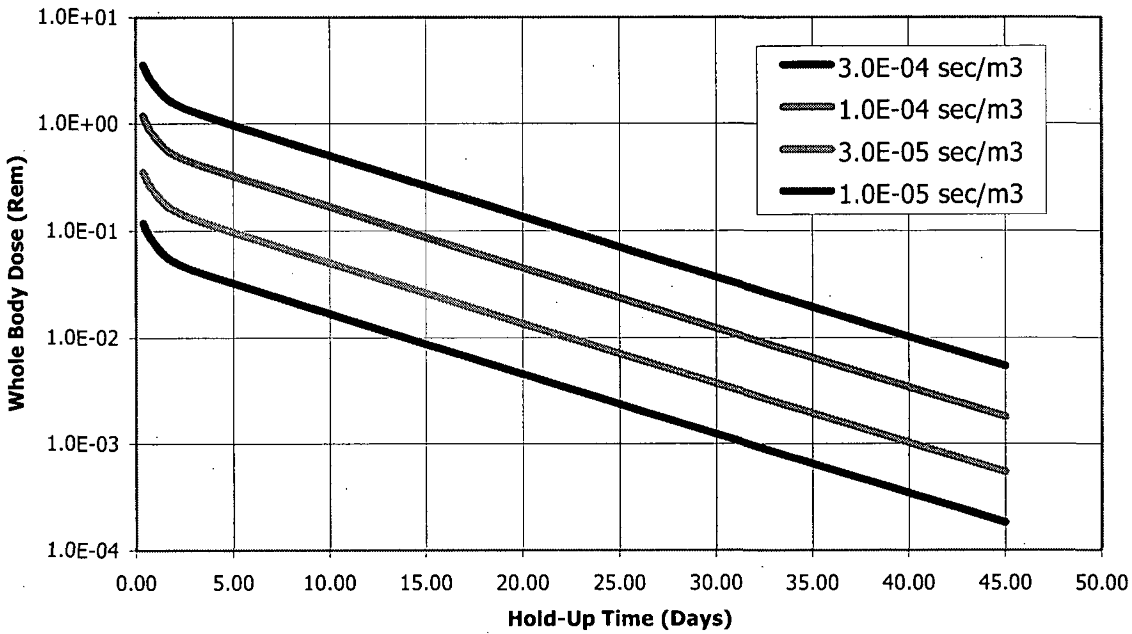

Figure S-4 Scenario 2 Xenon Whole Body Dose with Respect to Charcoal Hold Up

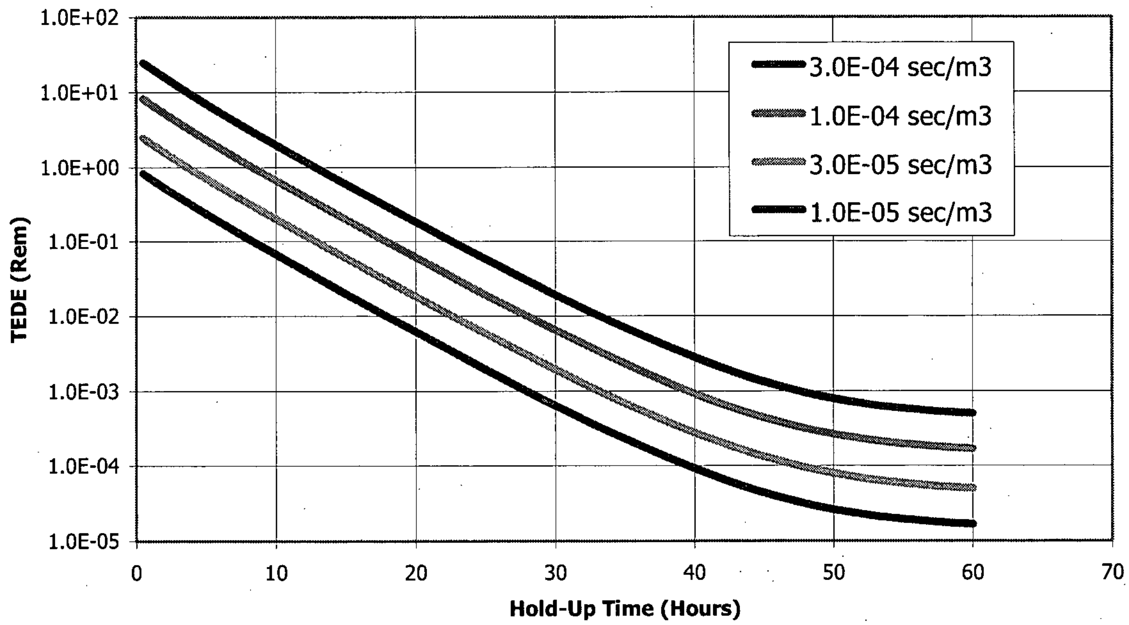

Figure S-5 Scenario 2 Krypton TEDE with Respect to Charcoal Hold Up Utilizing AST Methodology

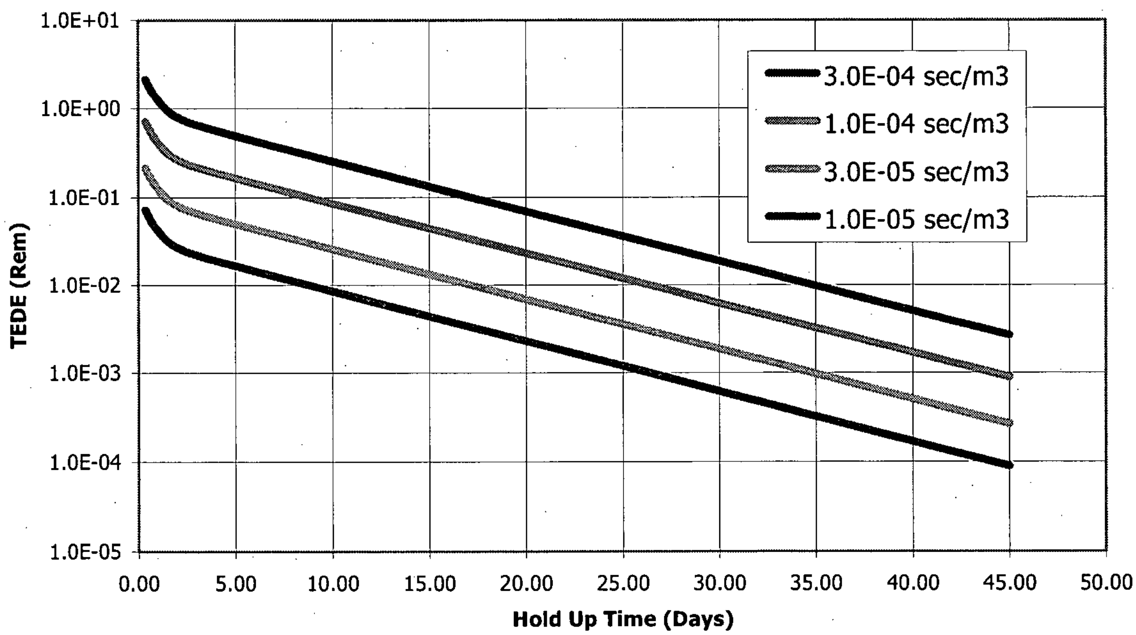

Figure S-6 Scenario 2 Xenon TEDE with Respect to Charcoal Hold Up Utilizing AST Methodology