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U. S. Nuclear Regulatory Commission
Washington, DC 20852–02738

Attention: Mel B. Fields

Subject: **GESTAR II Amendment 28 Revision 1, Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7**

Reference:
1. Letter to Document Control Desk from Margaret E. Harding, "GESTAR II Amendment 28, Misloaded Fuel Bundle Event Licensing Basis Change to Comply with Standard Review Plan 15.4.7," FLN–2004–009 dated May 17, 2004.
2. Letter to James F. Klapproth from Herbert N. Berkow, "Implementation of a Revised Review Process for Topical Reports," dated October 21, 2003.

GNF is hereby submitting this letter to revise the Reference 1 letter. An error was discovered in the execution of the code that analyzes the radiological offsite dose consequences; therefore, the only change being made is to correct the tables and figures contained in Attachment B, and any conclusions drawn from these tables and figures. These changes, in Appendices A and B only, in this electronic version was printed with all revision marks activated. Original text that has been deleted is marked as a ~~strikethrough~~ and new text is in color. For completeness, the entire package is being resubmitted in its original form. The page changes to GESTAR II are unaffected.

GNF hereby submits Amendment 28 to General Electric Standard Application for Reload Fuel (GESTAR II), NEDE–24011–P–A and NEDO–24011–A. This amendment proposes to make changes to GESTAR II and its U.S. Supplement to analyze the misloaded fuel bundle event as an *infrequent incident*. This change would bring this event to be in conformance with Standard Review Plan 15.4.7.

The misloaded fuel bundle event, because of its low probability of occurrence, has been classified as an accident (see the U.S. GESTAR II Section S.2.2.1). However, GNF continues to analyze this event (see the U.S. GESTAR II Sections S.2.2.3.6 and S.2.2.3.7) for each reload cycle as an anticipated operational occurrence (AOO) where it could possibly set the operating limit for a given plant cycle. In Amendment 28 to GESTAR II, GNF is not proposing to remove or reclassify the Misloaded fuel bundle event from its current classification as an accident, but it

is being proposed only to analyze this event in conformance with Standard Review Plan 15.4.7. A one time bounding radiological analysis is performed to show that the SRP 15.4.7 criteria are not exceeded. This event is shown to meet the established criteria; therefore, page changes are being proposed to GESTAR II and its U.S. Supplement that provides for analyzing the misloaded fuel bundle event in accordance with SRP 15.4.7.

Reference 2 requested that a need date be addressed in each submittal. Because Amendment 28 brings this misloaded fuel bundle event to be in conformance to already established NRC guidelines, it is believed that the NRC could complete their review within a three–month time frame, which would result in the amendment approval by the end of September 2004. This requested approval date is not required to support the reload analysis for any utility, but substantial reload analysis work will be saved when this amendment is approved.

The following information is attached to support the staff's review of this change:

Attachment A: Misloaded Fuel Bundle Event.

Attachment B: Bounding Radiological Analysis for a Misloaded Fuel Bundle Event.

Attachment C: GESTAR page changes proposed for Amendment 28. This electronic version was reprinted with all revision marks activated. Original text that has been deleted is marked as a ~~strikethrough~~ and new text is in color. Only the affected page changes are included. Both the proprietary version pages of GESTAR and the nonproprietary version pages of GESTAR are provided, however the proprietary version pages included for this change do not include any proprietary information.

The mislocated bundle event analysis will continue to be performed on potentially limiting plants until NRC concurrence with the GNF proposal to change the acceptance criteria for this event is received.

If you wish further information on the mislocated fuel bundle accident, please contact me or Jens Andersen, (910) 675–6083.

Sincerely,

Margaret E. Harding

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cc: F. Akstulewicz (NRC)
J. Wermiel (NRC)
J. F. Klapproth (GE)

A.0 Misloaded Fuel Bundle Event

A.1 Summary of Misloaded Fuel Bundle Event

GNF proposes to change the way that the analysis of the misloaded fuel bundle event is performed from that of an "incident of moderate frequency" category to that of an "infrequent incident" category. This would result in the misloaded fuel bundle event being evaluated at less demanding limits (10% of 10CFR100 limits versus 10CFR20 limits). An analysis change of the misloaded fuel bundle event will bring it in conformance with Standard Review Plan 15.4.7 (Reference A.1).

The adverse consequences from an incident of a misloaded fuel bundle event (either a mislocated fuel bundle or a misoriented fuel bundle) would be the failure of one or more fuel rods in a single fuel bundle that is operating in a higher–than–normal power range. The results of such an incident would be similar to a fuel bundle operating with one or more leaking fuel rods, "leakers." However, the radiological consequences, even though minor, would be difficult to assess for each fuel bundle in the core for each operating cycle. Therefore, in order to conservatively bound the consequences for this event, it is arbitrarily presumed that all of the fuel rods in the affected fuel bundle fail. To provide a clearly bounding analysis for this event, it is assumed that the four neighboring fuel bundles experience failure of all of their fuel rods—a total of all of the rods in five fuel bundles experience instantaneous failure during normal operation.

To further assure that the fuel bundles containing the maximum fission products for release are included, all five bundles (array independent) are multiplied by:

1. A factor of 1.4 to account for variations in fission product inventory over the operational cycle; and
2. A second factor of 2.5 to account for variations in cycle–dependent bundle power as a ratio to the end of cycle average bundle power.

Or a total factor of $1.4 \times 2.5 = 3.5$ to bound the bundle end of cycle inventory.

The radiological consequences of failing all of the fuel rods in a five–fuel bundle array has been analyzed for two different cases to provide results for different plant configurations: (1) those plants having a main steam line high radiation isolation trip, and (2) those plants without this trip. The two cases evaluated were (1) a regulatory analysis based upon similar accidents in which the plant is isolated and (2) a second case without isolation where the release is treated by an augmented offgas system.

The radiological consequences of failing all of the fuel rods in a five–fuel bundle array for the first case, plants having a main steam line high radiation isolation trip, show ~~that~~ the results ~~are well below the limit~~to be:

	Analysis Results	**Limit (from SRP 15.4.7–10% of 10CFR100.11)**
Whole body	~~0.31~~ 0.58 Rem	2.5 Rem
Thyroid	~~4.6~~30 Rem	30.0 Rem
~~Total effective dose equivalent (TEDE)~~	~~0.22 Rem~~	

For the second case, for plants not having a main steam line high radiation trip, the radiological consequences of failing all of the fuel rods in a five–fuel bundle array is dependent upon their long–term meterological parameters (Chi/Q, see Attachment B). Individual plants must verify periodically (once every few years) that they are within the limiting site meteorological criteria to ascertain that they are within the limits. The results of this meteorological conditions verification will be included for each plant during each reload analysis. Alternatively, individual plants may choose to determine that the misloaded fuel bundle event does not exceed the limits defined for anticipated operation occurrences. The results of either foregoing analysis will be reported in the supplemental reload licensing report.

No other adverse consequences will result from a misloaded fuel bundle as shown in Section A.5.

Therefore, the misloaded fuel bundle event will be ~~classified~~ analyzed as an infrequent incident in accordance with SRP 15.4.7 and the acceptance criteria are shown to be within the limit. GESTAR II Amendment 28 provides page changes to GESTAR II and its U. S. Supplement (Reference A.2) to reflect this classification of the misloaded fuel bundle event.

A.2 Misloaded Fuel Bundle Event Description

The event discussed in this report is the improper loading of a fuel bundle and subsequent operation of the core. Two types of misloading errors are possible, the mislocation of a fuel assembly and the misorientation of a fuel assembly. Three errors must occur for the mislocation event to take place. First, a bundle must be misloaded into a wrong position in the core. Second, the bundle that was supposed to be loaded where the mislocation occurred would have to be overlooked and also be placed in an incorrect location. Third, both misplaced bundles would have to be overlooked during the core verification performed following core loading. For the misorientation event, two errors must take place. First, the assembly must be rotated while being lowered into position. Second, the misoriented bundle would have to be overlooked during the core verification performed following the core loading.

Both the mislocated and the misoriented fuel bundle events are referred to as the misloaded fuel bundle event here for the purposes of discussion.

For a misloaded fuel bundle event, it is assumed that the improper loading of a fuel assembly is not discovered and corrected as a result of the core verification program, and the plant is operated throughout the operating cycle assuming that the design core configuration has been correctly implemented.

There is a strong possibility that the core monitoring system will recognize the misloaded fuel bundle, thereby allowing the reactor operators to mitigate the consequences of a misloaded fuel bundle. However, it is assumed that the misloaded fuel bundle is not monitored and that it operates through the cycle with fuel rods above the thermal–mechanical limit. The potential exists that one or more fuel rods will experience cladding failure. If this were to occur, the adverse consequences are detectable and can be suppressed during operation similar to leaking fuel rods resulting from other failure mechanisms. For the misloaded fuel bundle, the initial adverse consequences would consist of perforation of a small number of fuel rods in the assembly. Any perforations in the fuel cladding that may occur would be localized and not propagate to other assemblies. A control rod inserted in the vicinity of the leaking fuel rod(s) would suppress the power in the leaking fuel rod(s), consequently returning the thermal–hydraulic conditions to normal, and reducing the fission product release and the off–gas activity.

A.3 Event Classification and Rational for the Misloaded Fuel Bundle Event

A.3.1 Regulatory Guidance

The misloaded fuel bundle event is that incident listed in Table 15–1 of Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*. It comes under the general heading of Section 4 – "Reactivity and Power Distribution Anomalies" and the specific heading of Section 4.7 – "Inadvertent loading and operation of a fuel assembly in an improper position." General Design Criterion 10 requires that "...systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of Anticipated Operational Occurrences (AOOs)." NUREG–0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* describes the basis and acceptance criteria by which the USNRC evaluates events such as the misloaded fuel bundle event. NUREG–0800 is general and applies to all light water reactor designs. The NUREG–0800 Standard Review Plan for Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," lists two acceptance criteria:

1. To meet the requirements of GDC 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations.

2. In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 guidelines.

The requirements as applied by GNF have been specified in GESTAR II. GESTAR II describes the methodology and acceptance criteria to show that the outcome of events shown in Regulatory Guide 1.70 is acceptable as applied to BWRs. The unacceptable safety results for infrequent incidents (unexpected operational occurrences) listed in GESTAR (**referred to herein as criteria**) are as follows:

1. Release of radioactivity which results in dose consequences that exceed a small fraction (10%) of 10CFR100;
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.

A.3.2 Misloaded Fuel Bundle Event Acceptance Criteria

For the misloaded fuel bundle event, avoidance of unacceptable results related to infrequent incidents, criteria 1 and 2 is ensured primarily by showing that failing all of the rods in a five–bundle array does not exceed 10% of 10CFR100 offsite dose requirements.

The misloaded fuel bundle may exceed the operating mechanical LHGR limit, (i.e., the MAPLHGR limit) since it may have worse peaking than the normally–loaded bundle. If the misloaded fuel bundle operates above the operating mechanical LHGR limit, one or more rods may approach the design limit and experience cladding failure. If this were to occur, the adverse consequences would be the perforation of a small number of fuel rods in the misplaced bundle. The subsequent release of fission products to the reactor coolant would be detected by the off gas system.

It was found that the dosage rate for failure of all rods in a five–bundle cell (the number or array of the fuel rods does not matter) does not exceed 10% of 10CFR100 offsite dose requirements. Therefore, avoidance of unacceptable results related to infrequent incidents acceptance criteria 1 and 2 listed in Section A.3.1 are achieved~~ with large margin~~.

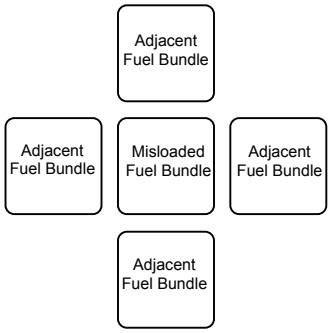
The misloaded fuel bundle event is not a limiting event for infrequent incidents acceptance criteria 3, 4 or 5, because the reactor remains at normal operating pressure throughout the event,

there is no challenge to the RPV and primary systems, such as process barrier stress limitations, ensuring compliance with criteria 3 and 5. Since SRVs do not open in this event, the containment also remains at normal operating pressure and temperature, ensuring compliance with criterion 4.

A.4 Misloaded Fuel Bundle Event Bounding Radiological Analysis

Analytical methods used to demonstrate that misloaded fuel bundle event meets the acceptance criteria requirements discussed in Section A.3.2 are summarized in Attachment B.

For a very conservative approach for the reactor core, a misloaded fuel bundle residing in a cell can be considered. Instead of one or two rods failing, for dosage rate considerations, it was assumed that all the fuel rods in a mislocated fuel assembly or a misoriented fuel assembly fail, and that all the rods in the adjacent fuel assemblies fail, see the included figure. With all rods in the five–bundle cell considered to fail the dosage rate will remain below 10% of the 10CFR100 offsite dose requirements as specified in SRP 15.7.4.



A.5 Misloaded Fuel Bundle Event Results and Conclusion

This section documents the results of the misloaded fuel bundle event analysis and how the acceptance criteria discussed in Section A.3.2 are met. The misloaded fuel bundle event analysis is based on initial conditions that yield the most severe results.

The radiological analysis performed for two different cases provides results for different plant configurations: (1) those plants having a main steam line high radiation isolation trip, and (2) those plants without this trip. For the first case, the results of the radiological analysis shows that the MSL high radiation trip maintains the off gas release to less that 10% of 10CFR100 providing that the very conservative Chi/Q value of 1.67×10^{-3} s/m^3 at 2 hours is met. For the second case, for plants not having a main steam line high radiation trip, the radiological consequences depend on the long–term meterological parameters (Chi/Q, see Attachment B). ~~These~~ All the plants will be subject to a periodic verification check to show that they are within the limiting site meteorological criteria to ascertain that they are within the limits.

For both plant configurations, the five criteria for infrequent incidents listed in Section A.3.1 are met for the misloaded fuel bundle event. The misloaded fuel bundle event was evaluated to meet the radioactive release limitations, as required. Compliance with criteria 1 and 2 is ensured primarily by showing that the failed rods are limited to less than 10% of 10CFR100. Because the reactor remains at normal operating pressure throughout the event, there is no challenge to the RPV and primary systems, such as process barrier stress limitations, ensuring compliance with criteria 3 and 5. Since no SRVs open, the containment also remains at normal operating pressure and temperature ensuring compliance with criterion 4.

A.6 References:

A.1. NUREG–0800, *Standard Review Plan*, Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," Draft Revision 2, April 1996.

A.2. *General Electric Standard Application for Reactor Fuel (GESTAR II)*, NEDE–24011–P–A–14, June 2000, and *General Electric Standard Application for Reactor Fuel (GESTAR II Supplement for United States)*, NEDE–24011–P–A–14–US, June 2000.

Misloaded Fuel Bundle Event Radiological Analyses for Offsite Dose

B.1 INTRODUCTION

Radiological evaluations were performed to address the potential offsite dose consequences associated with a misloaded fuel bundle event. No specific NRC guidance is provided in Standard Review Plan 15.4.7, "Inadvertent Loading And Operation Of A Fuel Assembly In An Improper Position," (Reference 1) as to acceptable methods for radiological analysis. Therefore, recourse was made to SRP 15.4.9 "Radiological Consequences of Control Rod Drop Accident (BWR)" (Reference 2) for guidance. The reference source term selected for these evaluations was based on NRC conservative assumptions for analysis of a design basis Control Rod Drop Accident.

Two alternate scenarios for the misloaded fuel bundle event (MFBE) were considered. The first followed the standard approach to analysis of the MFBE as outlined in U.S. NRC SRP 15.4.9. In this case, it was assumed that the fission product activity is airborne in the turbine and condenser following MSIV closure and leaks directly from the condenser to the atmosphere. In the second scenario, it was assumed that no automatic MSIV closure occurred in that the activity was transported to an augmented[1] offgas system. The release of this activity to the environment would be from the normal offgas release point after holdup in the treatment system. Calculations of post–accident doses for the Exclusion Area Boundary (EAB) were performed for each case to compare radiological consequences with the applicable exposure limits.

B.2 CONCLUSION

An analysis for a misloaded fuel bundle has been made assuming that all the rods in the misloaded fuel bundle plus all of the rods in the adjacent four bundles experience mechanical degradation resulting in fission gas release. To bound any potential variations in core and fuel two safety factors were applied: first a safety factor of 1.4 (see Table B-1) to account for variations in fission product inventory over the operational cycle; and second a safety factor of 2.5 was applied to account for variations in cycle dependent bundle power as a ratio to end of cycle average bundle power. Two cases were evaluated (1) a regulatory analysis based upon similar accidents in which the plant is isolated and (2) a second case without isolation where the release is treated by an augmented offgas system. The results of the first case using conservative siting parameters are ~~4.6~~30 Rem to the thyroid and ~~0.31~~ 0.58 Rem whole–body with a limiting Chi/Q of 1.67×10^{-3} s/m^3~~(0.22 Rem TEDE), which is a small fraction (~10%) of regulatory limits (the regulatory limits being 10% of 10CFR100.11)~~. The second case produced a family of curves to be applied on a case–by–case basis for plant specific offgas design.

[1] An augmented offgas system employs charcoal beds to hold up and delay the release of the non-condensable gas flow.

B.3 SOURCE TERM ASSUMPTIONS

Evaluation of both the design basis MFBE and the case where no MSIV closure occurs were based on the noble gas and iodine activity releases from the fuel which resulted from the following assumptions.

B.3.1 Fuel Damage

To obtain a bounding analysis, the MFBE was assumed to result in failure of the equivalent of five bundles (primary and four adjacent). No fuel melt was assumed to occur as a result of this event.

B.3.2 Fission Product Release from Fuel

Fission product released from the fuel is consistent with the provisions of SRP 15.4.9. The fission product inventory was based on previous long–term operation at full power with no allowance for decay prior to initiation of the event. To insure that the inventory is bounding upon such an event, a safety factor of 1.4 was generated which encompasses the variation in fission product inventory over the cycle of the operating fuel. For fuel that does not reach the melting temperature (all the involved fuel in this analysis), 10% of the noble gas inventory and 10% of the iodine inventory was assumed to be released to the coolant.

B.3.2.1 Bounding Assumption for Analysis

The calculations were performed with fission product inventories based on a power level of 5.75 MWt per bundle in the bundles that were assumed to fail. In addition, a safety factor of 2.5 (Reference 10) was used to insure that the peak bundle power to bundle average cycle power was bounded. Therefore, the bundle end of cycle inventory was multiplied by 1.4 x 2.5 = 3.5 to bound variations in bundle inventory.

B.4 SCENARIO EVALUATIONS

B.4.1 Analysis for MFBE with MSIV Closure (Scenario 1)

This analysis followed the SRP 15.4.9 conservative assumptions and utilized the source term previously described. This case assumes that the fission product activity is airborne in the main condenser.

B.4.1.1 Assumptions

The conservative analysis assumptions stated in SRP 15.4.9 provide that 100% of the noble gases and 10% of the iodines released to the coolant should be assumed to enter the steam and be transported to the condenser before MSIV closure. The 10% iodine fraction appears to be a very

conservative basis for partitioning of iodine between water and steam in the vessel since carryover values are typically less than 2% (Reference 3). Since the 10% iodine fraction and all of the noble gases are assumed to reach the condenser, no practical credit is taken for reduction of the available source term as a result of MSIV closure. A more realistic analysis of the transport can be reviewed on a case–by–case basis. The conservative transport assumptions have been used in the present analysis.

All of the noble gases reaching the condenser were assumed to remain airborne and available for leakage. Washout/plateout of 90% of the iodines reaching the condenser is assumed, with 10% remaining airborne and available for leakage.

All airborne activity in the condenser is assumed to leak from the condenser to the atmosphere at a rate of 1% per day. It is also assumed that the main condenser's mechanical vacuum pump is isolated.

No credit is taken for holdup and decay in the turbine building after release from the condenser. The release from the turbine building is assumed to occur at ground level, which was considered the worst predictable case for this event.

Thyroid dose conversion factors were taken from Regulatory Guide 1.109 (Reference 4) or Federal Guidance Report 11 (Reference 5) where Regulatory Guide 1.109 did not provide a value. Assumed breathing rates are from Regulatory Guide 1.3 (Reference 6). Whole–body gamma doses were based on a semi–infinite cloud calculation in accordance with Regulatory Guide 1.3 or Federal Guidance Report 12, Table 3.1 ~~for Total Effective Dose Equivalent (TEDE)~~ (Reference 7).

B.4.1.2 Methods of Analysis

The calculations were performed using methods derived from the CONAC04 computer program (Reference 8). The models and data used in CONAC04 are based on the Regulatory Guides or Standard Review Plans which define the NRC accepted methods and assumptions for evaluation of the accidents.

B.4.1.3 Results of Analysis

Activity inventories of nuclides that are airborne in the condenser are shown in Table ~~1~~B-2 for various times after initiation of the accident. The leakage rate from the condenser was 1% per day for all time periods. The corresponding time–integrated releases from the condenser to the environment are found in Table ~~2~~B-3.

The Chi/Q dispersion limit is back calculated from the 30 Rem thyroid dose based on the 2–hour Chi/Q at the EAB for a ground level release (though the total release integrated to 24 hours is used in the dose calculation). ~~Doses are calculated using an enveloping value of 2.5 x 10^{-3} sec/m^3 for the 2–hour Chi/Q at the EAB for a ground level release.~~The resultant Chi/Q dispersion

value was found to be 1.67×10^{-3} s/m^3, which bounds BWRs surveyed in Reference 9. Note that the curve given in Figure B-1 is conservative in that the dispersion coefficient is constant over the 24 hour period of release and does not provide for reductions in dose for longer term dispersion factors. ~~This value was determined in past studies to be in excess of all such values provided by BWR owners (Reference 9) in response to a request for site-specific data. The 2-hour doses obtained with the Chi/Q value were 4.6 Rem to the thyroid and 0.31 Rem whole-body (0.22 Rem TEDE). Since the calculated dose is directly proportional to the value of Chi/Q, d~~Doses for any other Chi/Q value may be scaled directly from these results and will be lower for any site when time variations in meteorology are accounted for. The relationships between calculated thyroid and whole–body doses and Chi/Q are shown in Figure B-1~~. These curves may be used to read offsite doses corresponding to a site-specific Chi/Q~~.

The offsite dose criterion established by SRP 15.4.7 for this accident is that doses should be a small fraction of the 10 CFR Part 100 guidelines; i.e., that the thyroid dose should be less than 30 Rem and the whole–body dose should be less than 2.5 Rem. Consequently, any dispersion coefficient less than the limiting value of 1.67×10^{-3} s/m^3 will result in doses less then the regulatory limit. Whole body and TEDE doses are a fraction of this limit~~the previously identified doses, based on a Chi/Q value which enveloped available data, were only 15% and 12% of the limit for thyroid dose and whole-body dose, respectively~~.

B.4.2 Analysis for MFBE without MSIV Closure (Scenario 2)

The method of analysis and assumptions, other than the release path assumptions discussed above, were consistent with the standard MFBE analysis described in the previous section. In Scenario 2 it was assumed that the MSIVs did not close immediately after initiation of the accident and that steam flow continued for some period of time. If sufficient reactor power is available for steam jet air ejector (SJAE) operation, some or all of the available activity is transported to the augmented offgas system. It was assumed that the activity processed by the treatment system would be released from the normal offgas release point after some holdup time in the system.

B.4.2.1 Assumptions

The available noble gas source term for the analysis is the same as that assumed in the analysis for the MFBE with MSIV closure; i.e., 10~~0~~% of the activity released from the fuel and 100% transported to the offgas system. For the purpose of this scenario, the entire MFBE noble gas source term was assumed to be released via the augmented offgas treatment system path to permit direct comparisons with doses calculated for the normal design basis MFBE condenser leakage path.

It was assumed that the iodine activity transported to the augmented offgas system was retained indefinitely and did not contribute to offsite doses. It might be argued that some iodine activity transport to the condenser, in addition to the conservatively assumed initial 10% fraction of the total fuel release, could occur because of continued iodine carryover with steam through the open

MSIVs. No offsite dose impact is expected, however, if the activity is being removed to the augmented offgas system by the air ejector.

If the event occurs at low power without the SJAE operating, the additional iodine activity due to carryover would not be expected to be significant. ~~(Note:~~ For Scenario 2 at low power levels without the SJAE operating, the offsite dose impact for noble gases is equivalent to Scenario 1, since 100% of the noble gas is assumed to be transported to the condenser.~~)~~ For example, if a carryover factor of 0.02 (ratio of microcuries of iodine per gram of steam to microcuries of iodine per gram of water) is assumed, the calculated iodine removal rate from the vessel at 5% of rated steam flow is in the approximate range of only 0.03 to 0.05% per minute. Moreover, the NRC assumption that 10% of the released iodines are instantaneously transported to the condenser is sufficiently conservative to bound the integrated carryover during the shutdown transient. For extended release periods (in excess of 3 hours), appropriate guidance should be provided for the operator to limit offsite releases.

B.4.2.2 Results of Analysis

Dose calculations were performed for an assumed release of ~~100~~10% of the kryptons and for an assumed release of ~~100~~10% of the xenons. Results of each of these calculations were plotted against an assumed delay time before release and shown in Figures B-2 ~~through 5~~and B-3. In Scenario 2 it was assumed that all of the remaining activity of each gas was released at approximately the same time. The doses shown in Figures B-2 ~~through 5~~and B-3 are integrated doses subsequent to release from the augmented offgas system. The 2–hour Exclusion Area boundary Chi/Q value applicable to the augmented offgas system release point is 3×10^{-4} sec/m^3. The uppermost curves shown in Figures B-2 ~~through 5~~and B-3 correspond to this value. Offsite doses for Chi/Q values not shown may be obtained by scaling directly from any of the curves, since the calculated dose is proportional to the Chi/Q value.

One method for establishing a conservative lower limit on the holdup time provided by the charcoal system would be to assume that the offgas system continues to operate after the MFBE, and that the condenser air inleakage rate which applies at rated conditions continues undiminished. The noble gas holdup times appropriate for normal operation would be applicable under these assumptions. This approach, however, ignores the potential for any operator control and the fact that the plant would be shutdown by other systems. In reality, interruption of system operation will increase the holdup time in the charcoal.

With the augmented offgas treatment systems that are presently in use, substantial decay times are assured for noble gases, and any iodine releases are negligible because of retention in the charcoal beds. The delay time in the charcoal beds is proportional to the mass of charcoal and to the dynamic adsorption coefficient for the gas (which is a function of operational temperature and humidity conditions in the charcoal) and inversely proportional to the condenser air inleakage flow rate. As a specific example, low temperature offgas systems supplied by GE provide minimum decay times of 46 hours for kryptons and 42 days for xenons, with the relatively high design basis air inleakage rate of 30 cubic feet per minute. For these decay times,

the doses corresponding to 100% release from Figures B-2 and B-3 for the enveloping Chi/Q value of 3×10^{-4} are approximately 1.6×10^{-3} and 7.9×10^{-3} ~~and 4.0×10^{-2} Rem~~ for kryptons and xenons, respectively. Summing these doses results in an approximate total of ~~4.1×10^{-2}~~ 9.5×10^{-3} Rem.

Delay times in ambient temperature charcoal offgas systems depend on plant–specific design conditions, but, in general, are shorter than delay times in low temperature systems. If, for example, it is assumed that such a system provides a decay time of 20 hours for kryptons and a typical factor of about 18 is used for the ratio between xenon holdup and krypton holdup (Reference 3), the corresponding xenon decay time would be about 360 hours or 15 days. Reading the curves in Figures B-2 and B-3 for Chi/Q = 3×10^{-4} results in doses of 1.21 Rem for kryptons and 1.30 Rem for xenons or a sum of 2.51 Rem which is roughly the 2.5 Rem dose limit.

B.5 DELAYED AUGMENTED OFFGAS SYSTEM OPERATION

In some instances, particularly early vintage BWRs, plant operating procedures allow continued bypassing of the offgas treatment system until late in power ascension. This operating mode is acceptable, provided the offgas radiation monitors (pre–treatment and/or post–treatment) are being utilized to automatically isolate the offgas treatment bypass line and/or offgas process line before the acceptable release rates are exceeded. Typically, the pretreatment monitor is included in plant Technical Specifications with requirements for periodic calibration and functional testing.

It is assumed that if the plant is at power and is bypassing its treatment system, the radioactive offgas release rate is therefore acceptable at that time. If, however, a subsequent condition develops such that excessive radioactivity is released from the reactor core, then the radiation detectors monitoring the offgas process line have sufficient sensitivity to indicate that isolation of either the bypass or process line should be initiated. (Note: For plants that do not have the capability to bypass the treatment system, the additional requirement of automatic isolation of the process line is not applicable.) The actual speed of response of these detectors is a function of the manufacturer and the method by which a continuous sample is obtained. Nevertheless, the selection of alarms and trip actions (including timer delays) for offgas radiation monitors, according to NRC guidance, is based on a correlation of acceptable offsite doses with system sensitivity to assure compliance with a plant' s Technical Specifications.

B.6 REFERENCES

1. U.S. NRC Standard Review Plan, Section 15.4.7, NUREG–0800, July 1981.

2. U.S. NRC Standard Review Plan, Section 15.4.9, NUREG–0800, July 1981.

3. NUREG–0016, Rev. 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors," January 1971.

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8. NEDO–32708, "Radiological Accident Evaluation — The "CONAC04A Code," August 1997.

9. NEDO–31400A, "Safety Evaluation for Eliminating The Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and SCRAM Function of the Main Steam Line Radiation Monitor," Class I, October 1992.

10. Letter, February 4, 2004, "Peaking Factor of 2.5, Justification and Approval," GE eDRF 0000–0006–0282.

Table B-1 Analysis Parameters

Analysis Parameter	Case 1 – Isolation	Case 2 – No Isolation
Inventory Parameters		
Bundle Power	5.75 MWt	5.75 MWt
Radial Peaking Factor	2.5	2.5
Safety Factor	1.4	1.4
Release Parameters		
Bundles Breached	5	5
Gap Fractions		
Noble Gas	10%	10%
Iodine	10%	10%
Fraction of Gap Released to coolant		
Noble Gas	100%	100%
Iodine	100%	100%
Fraction of fuel exceeding 2842°C	0%	0%
Transport to Turbine Condensers		
Noble Gas	100%	100%
Iodines	10%	10%
Condenser Removal and Transport		
Removal Fractions		
Noble Gas	0%	
Iodines	90%	
Leakage Rate	1% per day	
Period of Leakage	24 hours	
Offgas Removal and Transport		Iodines Removed. Noble Gas decay parameterized with Xe K_d = 18x Kr K_d
Meteorology and Dose Conversion Coefficients		
Dispersion Coefficient	1.67×10^{-3} s/m^3	Parameterized
Dose Conversion Factors	R.G. 1.109	R.G. 1.109
Breathing Rate	3.7×10^{-4} m^3/s	n/a

Table ~~1~~ B-2 Case 1 Inventory In Condenser in Curies

	Time into event in Hours						
	0	0.5	1	2	8	16	24
I-128	3.3E+01	1.4E+01	6.2E+00	1.2E+00	5.4E-05	8.9E-11	1.5E-16
I-129	1.0E-04	1.0E-04	1.0E-04	1.0E-04	1.0E-04	1.0E-04	9.9E-05
I-130	8.6E+01	8.4E+01	8.1E+01	7.7E+01	5.5E+01	3.5E+01	2.2E+01
I-131	2.6E+03	2.6E+03	2.6E+03	2.6E+03	2.6E+03	2.5E+03	2.4E+03
I-132	3.8E+03	3.3E+03	2.8E+03	2.1E+03	3.4E+02	3.0E+01	2.6E+00
I-133	5.4E+03	5.3E+03	5.2E+03	5.1E+03	4.1E+03	3.2E+03	2.4E+03
I-134	6.0E+03	4.0E+03	2.7E+03	1.2E+03	1.1E+01	1.9E-02	3.4E-05
I-135	5.1E+03	4.8E+03	4.6E+03	4.1E+03	2.2E+03	9.3E+02	4.0E+02
Kr-83m	3.4E+04	2.8E+04	2.3E+04	1.6E+04	1.7E+03	8.6E+01	4.4E+00
Kr-85	3.3E+03	3.3E+03	3.3E+03	3.3E+03	3.3E+03	3.3E+03	3.3E+03
Kr-85m	7.2E+04	6.6E+04	6.1E+04	5.3E+04	2.1E+04	6.0E+03	1.7E+03
Kr-87	1.4E+05	1.1E+05	8.0E+04	4.6E+04	1.7E+03	2.2E+01	2.7E-01
Kr-88	1.9E+05	1.7E+05	1.5E+05	1.2E+05	2.7E+04	3.7E+03	5.1E+02
Kr-89	2.4E+05	3.3E+02	4.6E-01	8.8E-07	4.5E-41	8.4E-87	1.6E-132
Kr-90	2.4E+05	0.0	0.0	0.0	0.0	0.0	0.0
Kr-91	1.8E+05	0.0	0.0	0.0	0.0	0.0	0.0
Kr-92	8.5E+04	0.0	0.0	0.0	0.0	0.0	0.0
Xe-131m	2.9E+03	2.9E+03	2.9E+03	2.9E+03	2.9E+03	2.8E+03	2.7E+03
Xe-133	5.4E+05	5.4E+05	5.4E+05	5.4E+05	5.2E+05	4.9E+05	4.7E+05
Xe-133m	1.7E+04	1.7E+04	1.7E+04	1.6E+04	1.5E+04	1.4E+04	1.2E+04
Xe-135	1.8E+05	1.8E+05	1.7E+05	1.6E+05	1.0E+05	5.4E+04	3.0E+04
Xe-135m	1.1E+05	2.7E+04	7.0E+03	4.6E+02	3.8E-05	1.4E-14	4.9E-24
Xe-137	4.7E+05	2.1E+03	9.4E+00	1.9E-04	1.1E-32	2.6E-70	6.1E-108
Xe-138	4.5E+05	1.0E+05	2.4E+04	1.3E+03	3.0E-05	2.0E-15	1.3E-25
Xe-139	3.5E+05	0.0	0.0	0.0	0.0	0.0	0.0

Table ~~2~~ B-3 Case 1 Integrated Release to Environment in Hours

	Time into event in Hours					
	0.5	1	2	8	16	24
I-128	4.61E-03	6.61E-03	7.87E-03	8.16E-03	8.16E-03	8.16E-03
I-129	2.09E-08	4.18E-08	8.36E-08	3.34E-07	6.67E-07	9.98E-07
I-130	1.77E-02	3.49E-02	6.79E-02	2.31E-01	3.78E-01	4.72E-01
I-131	5.50E-01	1.10E+00	2.19E+00	8.66E+00	1.71E+01	2.52E+01
I-132	7.37E-01	1.37E+00	2.38E+00	4.77E+00	5.19E+00	5.22E+00
I-133	1.12E+00	2.22E+00	4.37E+00	1.58E+01	2.79E+01	3.71E+01
I-134	1.03E+00	1.72E+00	2.49E+00	3.13E+00	3.14E+00	3.14E+00
I-135	1.03E+00	2.00E+00	3.81E+00	1.14E+01	1.63E+01	1.84E+01
Kr-83m	6.40E+00	1.17E+01	1.98E+01	3.57E+01	3.75E+01	3.76E+01
Kr-85	6.97E-01	1.39E+00	2.79E+00	1.11E+01	2.22E+01	3.33E+01
Kr-85m	1.44E+01	2.77E+01	5.15E+01	1.37E+02	1.77E+02	1.88E+02
Kr-87	2.52E+01	4.44E+01	7.01E+01	1.04E+02	1.05E+02	1.05E+02
Kr-88	3.82E+01	7.19E+01	1.28E+02	2.83E+02	3.21E+02	3.27E+02
Kr-89	7.55E+00	7.56E+00	7.56E+00	7.56E+00	7.56E+00	7.56E+00
Kr-90	1.27E+00	1.27E+00	1.27E+00	1.27E+00	1.27E+00	1.27E+00
Kr-91	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01	2.54E-01
Kr-92	2.60E-02	2.60E-02	2.60E-02	2.60E-02	2.60E-02	2.60E-02
Xe-131m	6.11E-01	1.22E+00	2.44E+00	9.68E+00	1.91E+01	2.84E+01
Xe-133	1.13E+02	2.25E+02	4.50E+02	1.77E+03	3.45E+03	5.06E+03
Xe-133m	3.50E+00	6.98E+00	1.39E+01	5.33E+01	1.01E+02	1.44E+02
Xe-135	3.75E+01	7.37E+01	1.42E+02	4.58E+02	7.08E+02	8.44E+02
Xe-135m	1.20E+01	1.51E+01	1.61E+01	1.62E+01	1.62E+01	1.62E+01
Xe-137	1.81E+01	1.82E+01	1.82E+01	1.82E+01	1.82E+01	1.82E+01
Xe-138	4.92E+01	6.06E+01	6.38E+01	6.40E+01	6.40E+01	6.40E+01
Xe-139	2.38E+00	2.38E+00	2.38E+00	2.38E+00	2.38E+00	2.38E+00

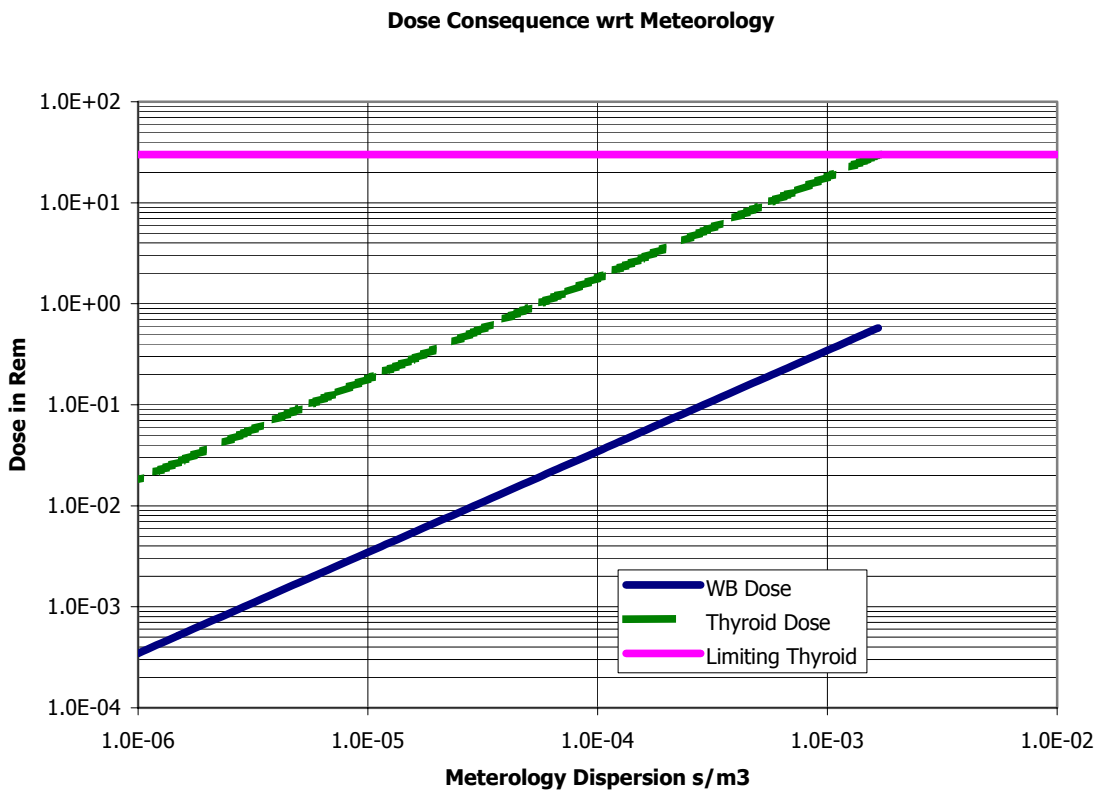

Figure B-1 Scenario 1 Dose with respect to Dispersion Coefficient

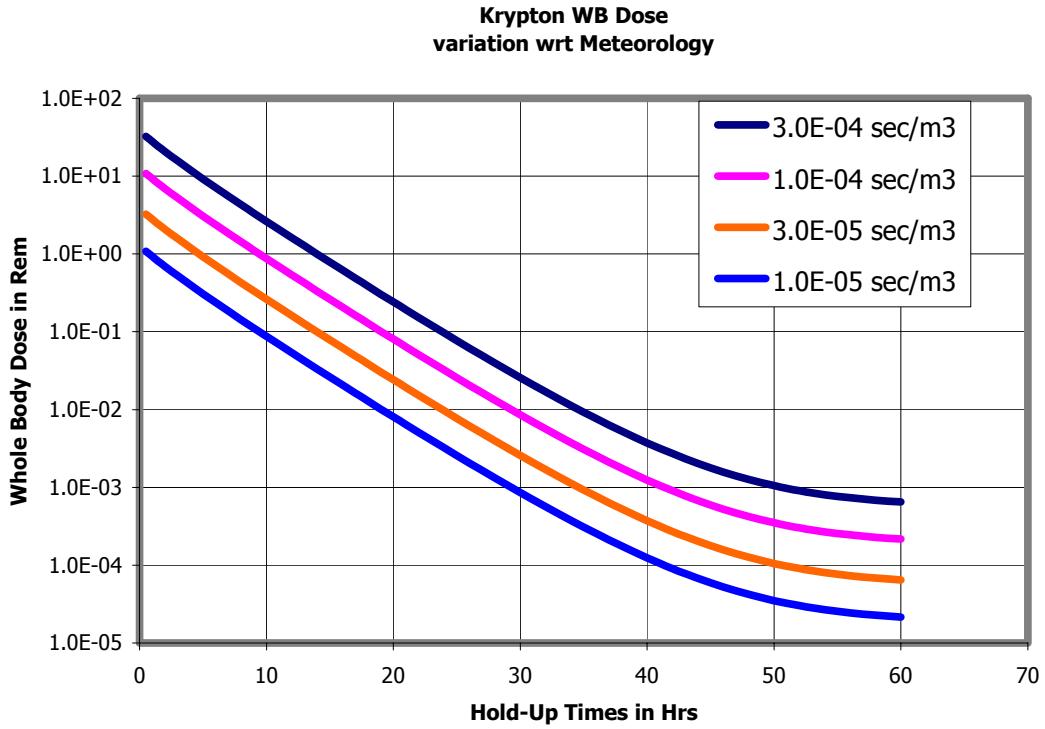

Figure B-2 Scenario 2 Krypton Whole Body Dose with Respect to Charcoal Hold Up

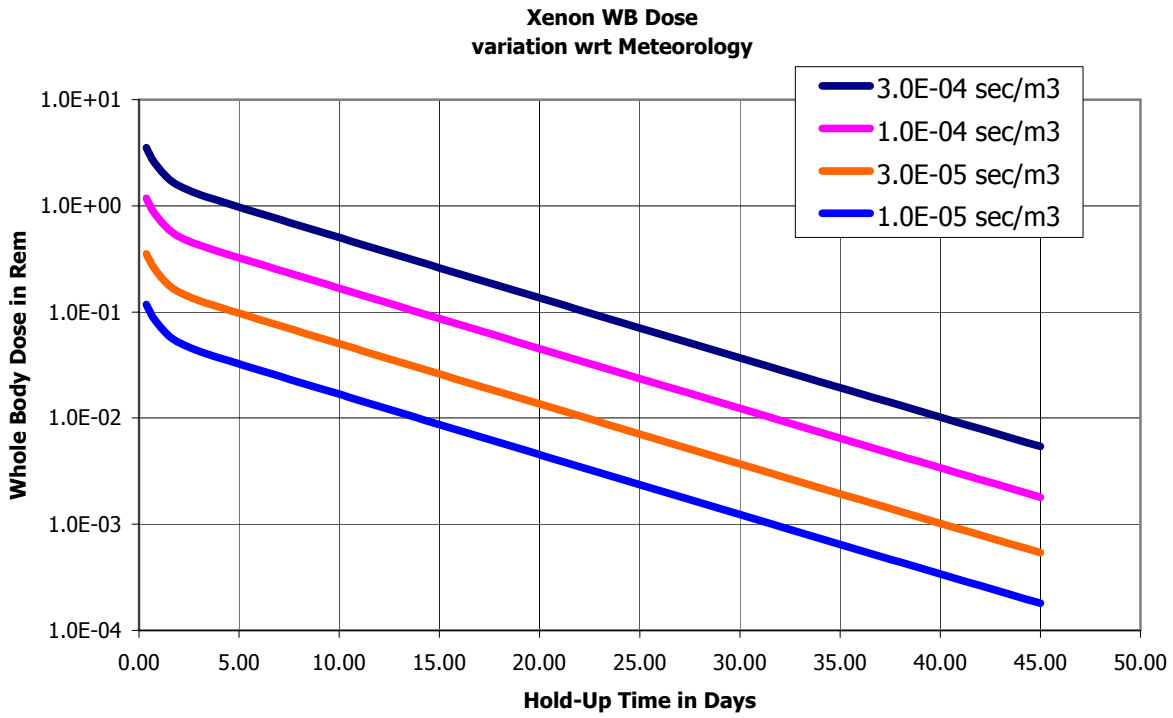

Figure B-3 Scenario 2 ~~Krypton TEDE~~Xenon Whole Body Dose with Respect to Charcoal Hold Up

GESTAR II Page Changes

- ~~For the Proprietary Version of GESTAR.~~
- For the Nonproprietary Version of GESTAR.

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

NEDO–24011–A–16
September 2004

Global Nuclear Fuels – Americas

Licensing Topical Report

General Electric Standard Application for Reactor Fuel

Approved: _____
Margaret E. Harding, Manager
Fuel Engineering Services

GESTAR II

Revision Status Sheet

Revision No.[a]	Amend. No.	GE Amend. Reference[b]	NRC Approval Reference[b]	Amendment Content
5	5	1 & 2	3	Administrative.
6	6	4	5	Incorporation of barrier clad.
7	7	6	7	GESTR–M fuel mechanical code application.
	8	8	9	Generic stability approach.
	9	10	11	Administrative.
8	10	12	13 & 14	Incorporation of GE8x8E and GE8x8EB fuel designs.
	11	15	16 & 17	Revised ODYN (GEMINI methods).
	12	18	19	Generic CRDA analysis for group notch plants utilizing BPWS.
	13	20	21	Administrative.
9	14	22	23	Safety limit MCPR reduction.
	15	24 & 25	26 & 27	GEXL–Plus correlation.
	16	28	29	Administrative.
	17	30	31	Modified requirements for rod pattern control system.
	18	32	33	Incorporation of GE8x8NB fuel design.
	19	34	35	Changes to technical specifications for power distribution limits.
10	21	36	37	Incorporation of GE8x8NB–1, –2, and –3 fuel designs.
	22	38 & 39	40	Fuel licensing acceptance criteria.
		41	42	Fuel channel bow effect on thermal margins.
11		43	44	TVAPS added and refueling accident updated to incorporate GE11 and later fuel designs.

[a] Only approved amendment are incorporated into a revision to GESTAR. Usually several approved amendments are combined together and incorporated into the document as one revision. GESTAR II began with Revision 4.

[b] See following pages for references.

Revision No.[a]	Amend. No.	GE Amend. Reference[b]	NRC Approval Reference[b]	Amendment Content
12			45	Superceded by Revision 13.
13		46	47	Cold water event determination.
14	25	48	49	Cycle–specific Safety Limit MCPR.
	26	50	51, 52	Administrative including approvals of Stability and ATWS, inclusion of MLHGR, classifying PRDF, implementing improved GE steady–state methods, et. al.
15	27			Administrative including approvals of LOCA's SAFER/GESTR, Upper Bound PCT, and TASC; and TRACG. Also, clarified position on stability. Several errors corrected on reload licensing. Other minor errors cleaned up, et. al.
16	28			Misloaded fuel bundle event analysis changed to conform to SRP 15.4.7.

References

(Revision Status Sheet)

1. Letter from J. S. Charnley (GE) to R. L. Tedesco (NRC), *Administrative Amendments to GESTAR II*, August 31, 1982.

2. Letter from J. S. Charnley (GE) to T. Novak (NRC), *Additional Administrative Meetings Amendment to NEDE–24011–P–A–4*, September 8, 1982.

3. Letter from C. O. Thomas (NRC) to G. G. Sherwood (GE), *Administrative Amendment to GE LTR NEDE–24011–P–A and NEDE–24011–P–A–4–US*, November 29, 1982.

4. Letter from J. S. Charnley (GE) to F. J. Miraglia (NRC), *Barrier Fuel Amendment to NEDE–24011–P–A*, November 19, 1982.

5. Letter from C. O. Thomas (NRC) to J. S. Charnley (NRC), *Acceptance for Referencing of Licensing Topical Report NEDE 24011–P–A Barrier Fuel Amendment to GE Standard Application for Reactor Fuel*, April 13, 1983.

Discussions of plant specific and generic rod drop accident evaluation methodologies are presented in the country–specific supplement to this base document.

1.1.12 Refueling Accident Analysis

A. The consequences of a refueling accident as presented in the country–specific supplement to this base document or the plant FSAR shall be confirmed as bounding or a new analysis shall be performed (using the methods and assumptions described in the country supplement) and documented when a new fuel design is introduced.

1.1.13 Anticipated Transient Without Scram

The fuel must meet either criteria A or B below:

A. A negative core moderator void reactivity coefficient, consistent with the analyzed range of void coefficients provided in References 1–7 and 1–8, shall be maintained for any operating conditions above the startup critical condition.

B. If criterion 1.1.13.A is not satisfied, the limiting events (as described in References 1–7 and 1–8) will be evaluated to demonstrate that the plant response is within the ATWS criteria specified in References 1–7 and 1–8.

1.1.14 Misloaded Fuel Bundle Event Analysis

A. The mislocated and misoriented bundle loading error events are subject to the radiological limits of 10% of 10CFR100. Bounding radiological analysis of these events is referenced in the country–specific supplement to this base document. Individual plants must verify periodically (once every few years) that they are within the limiting site meteorological criteria to ascertain that they are within the limits, or they will be required to demonstrate no fuel failure occurs for these two events.

1.2 Basis for Fuel Licensing Criteria

The following provides the basis for the criteria documented in Subsection 1.1.

1.2.1 General Criteria

A. *NRC–approved analytical models and analysis procedures will be applied.*

Consistent with current practice, NRC–approved procedures and methods are used to evaluate new fuel designs.

B. *New design features will be included in lead use assemblies.*

GE's "test before use" fuel design philosophy includes irradiation experience with new fuel design features in full–scale fuel assemblies (Lead Use Assemblies) in operating reactors prior to standard reload application. A method for licensing LUAs and the NRC acceptance of this method are documented in References 1–9 and 1–10, respectively.

the core loading pattern, this limit is cycle dependent for each plant and is calculated just prior to operation of the cycle.

B. *For each new fuel design the applicability of generic MCPR analyses described in Section 4 or in the country–specific supplement to this base document shall be confirmed for each operating cycle or a plant–specific analysis will be performed.*

Generic event analysis results have been calculated for the Rod Withdrawal Error. These analyses are dependent upon the fuel design for BWR 3–5 plants without ARTS and the analytical methods, and must be reconfirmed whenever there is a change in either. Currently the generic analysis for these plants is approved for fuel designs through P8x8R and BP8x8R with both GENESIS and GEMINI methods and the GEXL and GEXL–PLUS critical power correlation. Analysis for these plants with GE8x8E/EB and GE8x8NB fuel must be performed on a cycle–specific basis. The generic analyses for plants with ARTS and BWR/6 plants with enrichments less than 3.25 weight percent enrichment are applicable to fuel designs through GE8x8E/EB with GENESIS and GEMINI methods and GEXL critical power correlation. A plant cycle specific evaluation must be performed for the GE8x8E/EB fuel design with GEXL–PLUS and the GE8x8NB fuel designs until a sufficient database exists to determine the applicability of the generic analyses. Similar cycle specific analyses will be performed for new fuel designs until an adequate database exists to perform generic analyses using methods previously approved by the NRC.

~~The mislocated fuel loading error is performed for initial cores and reload cores where the resultant CPR response may establish the operating limit MCPR (OLMCPR).~~

1.2.7 Critical Power Correlation

A. *The currently approved critical power correlations will be confirmed or a new correlation will be established when there is a change in wetted parameters of the flow geometry; this specifically includes fuel and water rod diameter, channel sizing and spacer design.*

The coefficients for the critical power correlation of a fuel design will be determined generically based on the criteria documented in Subsection 1.1.7. The fuel design parameters given in these criteria are those that have the primary effect on determining the need for a new critical power correlation when there is a change in the fuel design. New coefficients for the critical power correlation will be provided in the fuel design information report.

B. *A new correlation may be established if significant new data exists for a fuel design(s).*

When significant new data have been taken for a fuel design, a better fit to the data may be achieved by adjusting the coefficients in the critical power correlation. The resulting new critical power correlation would be a more accurate representation of

The consequences of the refueling accident are primarily dependent upon the number of fuel rods in a bundle. When the number of fuel rods changes, the effect on the refueling accident must be generically determined based on approved NRC methods. The results of this analysis will be documented in the fuel design information report.

1.2.13 Anticipated Transient Without Scram

The fuel must meet either criteria A or B below.

This evaluation will assure compliance to the generic ATWS approval. Nuclear inputs used in the evaluation will be obtained from the generic nuclear analyses documented in Subsection 1.2.3.

A. *A negative core moderator void reactivity coefficient, consistent with the analyzed range of void coefficients provided in References 1–7 and 1–8 shall be maintained for any operating conditions above the startup critical condition.*

   In response to the requirements of Alternate 3, set forth in NUREG–0460, References 1–7 and 1–8 present assessments of the capabilities of representative BWR plants to mitigate the consequences of a postulated ATWS event. Sensitivity studies are provided for the key parameters affecting plant response during the most limiting events requiring ATWS consideration. Values of parameters that fall within the range of characteristics studied have been shown to satisfy the ATWS acceptance criteria.

   In terms of core response to an ATWS event, the core moderator void reactivity coefficient is the key parameter. Maintaining this coefficient within the range of point model void coefficients (or equivalent one–dimensional void coefficients) assumed in the sensitivity studies presented in References 1–7 and 1–8 when loading new fuel designs, assures that the conclusions reached regarding BWR mitigation of an ATWS event are still valid.

B. *If criterion 1.1.13 is not satisfied, the limiting events (as described in References 1–7 and 1–8) will be evaluated to demonstrate that the plant response is within the ATWS criteria specified in References 1–7 and 1–8.*

   For new fuel designs that have core moderator void reactivity coefficients outside the range of void coefficients assumed in the sensitivity studies presented in References 1–7 and 1–8, a specific evaluation will be performed. The most limiting events identified in References 1–7 and 1–8 will be evaluated to assure that core and plant response is within the documented ATWS acceptance criteria.

1.2.14 Misloaded Fuel Bundle Event Analysis

A. *The mislocated and misoriented bundle loading error events are subject to the radiological limits of 10% of 10CFR100. Bounding radiological analysis of these events is referenced in the country–specific supplement to this base document. Individual plants must verify periodically (once every few years) that they are within*

the limiting site meteorological criteria to ascertain that they are within the limits, or they will be required to demonstrate no fuel failure occurs for these two events.

The consequences of the misloaded fuel bundle event are primarily dependent upon each plant's long–term meteorological parameters. The results of this meteorological conditions verification will be included for each plant during each reload analysis. Alternatively, individual plants may choose to determine that the misloaded fuel bundle event does not exceed the limits defined for anticipated operation occurrences. The results of either foregoing analysis will be reported in the supplemental reload licensing report.

1.3 Core Configuration

Each BWR reactor core is comprised of core cells. Each core cell consists of a control rod and four fuel assemblies that immediately surround it (Figure 1–1). Each core cell is associated with a four–lobed fuel support piece. Around the outer edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel support pieces. The four fuel assemblies are lowered into the core cell and, when seated, springs mounted at the tops of the channels force the channels into the corners of the cell such that the sides of the channels contact the grid beams (Figure 1–2).

Core lattice designations are based upon relative water gap size between adjacent fuel assemblies and dimensional characteristics of the basic fuel assembly and channel. The core lattice descriptions and a definition of the specific type of lattice used for each plant are contained in Reference 1–2.

will be occurrences where the number and/or types of bundles in the reference design and the actual core loading do not agree exactly.

Any differences between the reference loading pattern and the actual loading pattern are evaluated as described in Section 3.4.

3.2.2 Power Distribution

The core power distribution is a function of fuel bundle design, core loading, control rod pattern, core exposure distributions and core coolant flow rate. The thermal performance parameters, MAPLHGR, MLHGR and MCPR (defined in Table 3–1), limit unacceptable core power distributions.

3.2.2.1 Power Distribution Measurements

The techniques for measurement of the power distribution within the reactor core, together with instrumentation correlations and operation limits, are discussed in Reference 3–1.

3.2.2.2 Power Distribution Accuracy

The accuracy of the calculated power distributions is discussed in References 3–4, 3–5, 3–16, 3–17 and 3–18.

3.2.2.3 Power Distribution Anomalies

Stringent inspection procedures are utilized to ensure the correct arrangement of the core following fuel loading. A fuel loading error (a mislocated or a misoriented fuel bundle in the core) ~~would~~ has been shown to be a very improbable event, but ~~calculations have been performed to determine the effects of such events on CPR~~ has been analyzed as an infrequent event in accordance with Standard Review Plan 15.4.7. Fuel loading error is discussed further in the country–specific supplement to this document.

The inherent design characteristics of the BWR are well suited to limit gross power tilting. The stabilizing nature of the large moderator void coefficient effectively reduces the effect of perturbations on the power distribution. In addition, the in–core instrumentation system, together with the on–line computer, provides the operator with prompt information on the power distribution so that he can readily use control rods or other means to limit the undesirable effects of power tilting. Because of these design characteristics, it is not necessary to allocate a specific margin in the peaking factor to account for power tilt. If, for some reason, the power distribution could not be maintained within normal limits using control rods and flow, then the total core power would have to be reduced.

3.2.3 Reactivity Coefficients

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in calculating stability and evaluating the response of the core to external disturbances. The base initial condition of the system and the postulated initiating

In the few instances where fuel bundles near the edge are quadrant–loaded asymmetrically, the error induced by reflecting real readings is partially negated by the fuel type dependent correlations. Any remaining error is considered to be of negligible second order. Further, because such bundles are in low power regions, it is highly unlikely that one of them is a limiting bundle.

In the rare case of the reactor being operated with an asymmetric control rod pattern, the reflection of real string readings is not utilized. In this instance, readings at locations without strings are inferred by interpolation of the real string values in the immediate vicinity.

3.4.2.9 Shutdown Margin

The cold shutdown margin is always recalculated for the final core loading. Adequate shutdown margin is verified experimentally during the startup.

3.4.3 Re–Examination of Bases

If the final loading plan does not meet the criteria of Subsection 3.4.2, a re–examination of the parameters that determine the operating limits is performed. Based on results of the sensitivity studies of the operating limits to these parameters, conservative bounds have been set on the allowable change from the reference. These parameters are:

1. Scram reactivity insertion.
2. Dynamic void coefficient.
3. Peak fuel enthalpy during rod drop accident.
4. Cold shutdown margin.
5. Standby liquid control system shutdown margin.

~~**6.**Change in critical power ratio due to a misloaded fuel assembly.~~

~~7.~~6.Rod block monitor response to a rod withdrawal error.

~~8.~~7.Safety Limit MCPR.

These parameters were chosen by one of the following two criteria:

(1) It is a parameter whose magnitude or behavior is explicitly reported in the supplemental reload licensing report.

Examples:

Cold shutdown margin, peak fuel enthalpy in Rod Drop Accident, ~~change in CPR due to a misloaded assembly,~~ and Rod Block Monitor response.

(2) It is a parameter important to the quantification of an operating limit.

Examples:

Scram reactivity insertion and dynamic void coefficient affect the operating limit MCPR.

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

NEDO–24011–A–16–US
September 2004

Licensing Topical Report

General Electric Standard Application for Reactor Fuel

(Supplement for United States)

Approved: _____

Margaret E. Harding, Manager
Fuel Engineering Services

Contents

Appendices

(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, Unexpected Operational Occurrences 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 accident conditions which, because of lower frequency of occurrence, do not strictly qualify as moderately frequent incidents, but are evaluated to meet the fuel cladding integrity safety limit MCPR, are the mislocated bundle accident and the misoriented bundle accident. Descriptions of these events are given in S.2.2.3.~~

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.

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.2.2.2 Unexpected Operational Occurrences (Infrequent Incidents)

The criteria for the infrequent events allow for calculated failures in a small fraction of fuel elements. ~~Detailed fuel integrity discussions that show no fuel failures have been provided to the NRC generically in Reference S–8.~~

The mislocated and misoriented bundle loading error events are discussed in this category even though they are considered, because of their low probability of occurrence, as limiting faults. Evaluation of BWR operating experience demonstrates that the actual frequency of occurrence of several other ~~of the~~ events currently categorized as moderate frequency events in Subsection S.2.2.1 is less than once in 20 years. However, these other events are currently analyzed, and corresponding limitations defined, as if they were moderately frequent events. ~~Therefore, there are no operational occurrence analyses performed by GE that fall in the infrequent event category.~~

The mislocated and misoriented bundle loading error events are subject to the radiological limits of 10% of 10CFR100. Bounding radiological analysis of these events is contained in Reference S–99. Individual plants must verify periodically (once every few years) that they are within the limiting site meteorological criteria contained in Reference S–99 to ascertain that they are within the limits, or they will be required to demonstrate no fuel failure occurs for these two events, i.e., analyze these events as AOOs (see References S–45 and S–46). Individual utilities may elect to substitute an alternative approach as noted in Reference S–47 for this limit. The method chosen and the results of the mislocated and/or misoriented fuel bundle analysis will be reported in the supplemental reload licensing report (see Appendix A).

S.2.2.2.1 Mislocated Bundle Accident

The mislocated fuel bundle 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.

There is a strong possibility that the core monitor will recognize the mislocated fuel bundle, thereby allowing the reactor operators to mitigate the consequences of this event. In the best situation 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. A less effective situation is where the mislocated bundle has a bundle between it and an instrument. An ineffective situation occurs when the core monitor will not recognize the mislocation if the monitoring system is not radially TIP or LPRM adapted.

Assuming the mislocated bundle is not monitored, one possible state of operation for the fuel bundle is that it operates through the cycle close to or above the fuel thermal mechanical limit. The potential exists that if the fuel bundle operates above the thermal mechanical limit, one or more fuel rods may experience cladding failure. If this were to occur, the adverse consequence of operation are detectable and can be suppressed during operation just like leaking fuel rods resulting from other causes. In this context, the adverse consequence is the perforation of a small number of fuel rods in the mislocated fuel assembly. Any perforations that may result would be localized, there would be only a few perforations, and the perforations would not propagate to other fuel rods or fuel assemblies. The perforation of a small number of fuel rods leads to the release of fission products to the reactor coolant, which are detected by the offgas system. A control rod inserted in the vicinity of the leaking fuel rods suppresses the power in the leaking fuel rods, returns the thermal–hydraulic condition to normal heat transfer with its characteristic low temperature difference between the cladding and the coolant, and reduces the fission product release and offgas.

Further discussion on the analysis methods for the mislocated bundle accident is given in Reference S–99.

Proper location of the fuel assembly in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. GE provides recommended fuel assembly loading instructions for the initial core as part of the Startup Test Instructions (STIs). It is expected that the plant owners use similar procedures during subsequent refueling operations. 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.

S.2.2.2.2 Misoriented Bundle Accident

The misoriented bundle accident has been evaluated on a generic bounding basis.

Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and assured by verification 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.

Experience has demonstrated that these design features are clearly visible so that any misoriented fuel assembly would be readily identifiable during core loading verification.

The bounding analysis for the misoriented fuel assembly is discussed in detail in Reference S–99. Approval of these methods is given in Reference S–46. Individual utilities may elect to substitute an alternative approach as noted in Reference S–47 for this limit.

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. ~~The mislocated and misoriented bundle loading error events are also discussed in this category even though they are required to meet the safety limit MCPR criteria discussed in Subsection S.2.2.1 for the AOO events.~~

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

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

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.2.2.3.6 Mislocated Bundle Accident**~~

~~Analysis of the mislocated bundle accident is performed for initial cores and reload cores where the resultant CPR response may establish the operating limit MCPR (OLMCPR).~~

~~**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.~~

~~Since 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. If the 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 violates the Tech Spec operating limit MCPR.~~

~~**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.~~

~~There is a strong possibility that the core monitor will recognize and mitigate the consequences of a mislocated bundle. In the best situation 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. A less effective situation is where the mislocated bundle has a bundle between it and an instrument. An ineffective situation occurs when the core monitor will not recognize the mislocation if the monitoring system is not radially TIP or LPRM adapted.~~

~~Assuming the mislocated bundle is not monitored, one possible state of operation for the fuel bundle is that it operates through the cycle close to or above the fuel thermal mechanical limit. The potential exists that if the fuel bundle operates above the thermal mechanical limit, one or more fuel rods may experience cladding failure. If this were to occur, the adverse consequence of operation are detectable and can be suppressed during operation just like leaking fuel rods resulting from other causes. In this context, the adverse consequence is the perforation of a small number of fuel rods in the mislocated fuel assembly. Any perforations that may result would be localized, there would be only a few perforations, and the perforations would not propagate to other fuel rods or fuel assemblies. The perforation of a small number of fuel rods leads to the release of fission products to the reactor coolant, which are detected by the offgas system. A control rod inserted in the vicinity of the leaking fuel rods suppresses the power in the leaking fuel rods, returns the thermal–hydraulic condition to normal heat transfer with its characteristic low temperature difference between the cladding and the coolant, and reduces the fission product release and offgas.~~

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

~~Proper location of the fuel assembly in the reactor core is readily verified by visual observation and assured by verification procedures during core loading. GE provides recommended fuel assembly loading instructions for the initial core as part of the Startup Test Instructions (STIs). It is expected that the plant owners use similar procedures during subsequent refueling operations. 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.~~

~~S.2.2.3.7 Misoriented Bundle Accident~~

~~The misoriented bundle accident is evaluated on a cycle specific basis.~~

~~Proper orientation of fuel assemblies in the reactor core is readily verified by visual observation and assured by verification 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.~~

~~Experience has demonstrated that these design features are clearly visible so that any misoriented fuel assembly would be readily identifiable during core loading verification.~~

~~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.2.3 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

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,~~ 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,~~ 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

S.6 References

S–1 *General Electric Standard Application for Reactor Fuel*, GESTAR II — Base Document (NEDE–24011–P–A, Revision 15, August 2003).

S–2 *General Electric Fuel Bundle Designs*, NEDE–31152–P, Revision 8, April 2001.

S–3 *General Electric Fuel Bundle Designs Evaluated with TEXICO/CLAM Analyses Bases*, latest version, NEDE–31151–P.

S–4 *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, Regulatory Guide 1.70, Revision 3, November 1978.

S–5 *GESSAR II–238; BWR/6 Nuclear Island Design*, NRC Docket No. STN50–447.

S–6 Letter from R. E. Engel (GE) to T. A. Ippolito, *Control Rod Withdrawal Error*, August 28, 1981.

S–7 Letter from L. S. Rubenstein (NRC) to R. E. Engel (GE), *Change in General Electric Analysis of Rod Withdrawal Error*, November 25, 1981.

S–8 ~~Letter from E. A. Hughes to R. C. DeYoung, *Turbine Trip without Bypass Analyzed as an Infrequent Incident*, October 5, 1976.~~not used.

S–9 Letter from J. S. Charnley (GE) to C. O. Thomas (NRC), *Licensing Credit for Banked Position Withdrawal Sequences on Group Notch Plants*, May 10, 1985.

S–10 Letter from T. Pickens (BWROG) to G. C. Lainas (NRC), *Amendment 17 to General Electric Licensing Topical Report NEDE–24011–P–A*, August 15, 1986.

S–11 C. J. Paone, *Banked Position Withdrawal Sequence*, January 1977 (NEDO–21231).

S–12 C. J. Paone and J. A. Woolley, *Rod Drop Accident Analysis for Large Boiling Water Reactors*, Licensing Topical Report, March 1972 (NEDO–10527).

S–13 R. C. Stirn, C. J. Paone, and R. M. Young, *Rod Drop Accident Analysis for Large Boiling Water Reactors*, Licensing Topical Report, July 1972 (NEDO–10527, Supplement 1).

S–14 R. C. Stirn, C. J. Paone, and J. M. Haun, Rod Drop Accident Analysis for Large Boiling Water Reactors, Addendum No. 2, Exposed Cores, Licensing Topical Report, January 1973 (NEDO–10527, Supplement 2).

S–15 Letter from R. E. Engel (GE) to D. M. Vassallo (NRC), *Elimination of Control Rod Drop Accident Analysis for Banked Position Withdrawal Sequence Plants*, February 24, 1982.

S–16 *Steady–State Nuclear Methods*, April 1985 (NEDE–30130–P–A and NEDO–30130–A).

S–97 *BWROG In–Service Pressure Relief Technical Specification Revision Licensing Topical Report*, NEDC–31753P, February 1990.

S–98 Letter from Ashok Thadani (USNRC) to Cynthia Tully (BWROG), "Acceptance for Referencing of Licensing Topical Report NEDC–31753P 'BWROG In–Service Pressure Relief Technical Specification Revision Licensing Topical Report' (TAC No. M79265)," March 8, 1993.

S–99 Letter from Margaret E. Harding (GNF) to Document Control Desk (Bo Pham, NRC), "GESTAR II Amendment 28, Misloaded Fuel Bundle Event Licensing Basis Change," (date).

S–100 Letter from xxxx (NRC) to xxxx (GNF), "xxxx …," (date).

11. Cycle MCPR Values

Safety limit: 1.nn

Single loop operation safety limit: 1.nn

Non–Pressurization Events:

Exposure Range: BOC to EOC	
	Appropriate Fuel Design(s)
Limiting coolant temperature decrease transient	1.nn
Fuel loading error (if reported as an AOO)	1.nn
Rod withdrawal error	1.nn

Pressurization Events:

Exposure range: BOC to EOC – n MWd/MT (n MWd/ST)		
	Option A Appropriate Fuel Design(s)	**Option B Appropriate Fuel Design(s)**
Limiting pressure and power increase event	1.nn	1.nn
Feedwater controller failure	1.nn	1.nn
Pressure regulator failure downscale (BWR 6)	1.nn	1.nn
Exposure range: EOC – n MWd/MT (n MWd/ST) to EOC		
Limiting pressure and power increase event	1.nn	1.nn
Feedwater controller failure	1.nn	1.nn
Pressure regulator failure downscale (BWR 6)	1.nn	1.nn

12. Overpressurization Analysis Summary

Transient	P_{sl} (psig)	P_v (psiv)	Plant Response
MSIV closure (flux scram)	nnnn	nnnn	Figure n

13. Loading Error Results [6]

Variable water gap misoriented bundle analysis: a

Event	ΔCPR
a	0.nnn

14. Control Rod Drop Analysis Results [7]

Plant specific analysis results:

Resultant peak enthalpy, cold:	nnn.n
Resultant peak enthalpy, HSB:	nnn.n

15. Stability Analysis Results

Plant stability option implementation should be stated, and reference made to NRC–approved licensing topical reports, including where appropriate, plant–specific stability reports. Plant specific analysis results or reconfirming analysis results for each cycle, where required, will be presented and applicable figures included.

[6] When radiological conditions have been verified to be within the limits presented in the GESTAR II U.S. Supplement, NEDE–24011–P–A–14–US, May 2000, analysis of the mislocated and/or misoriented fuel bundle analysis are not performed. NRC approval is documented in GESTAR II.

[7] For banked position withdrawal sequence plant an analysis of control rod drop need not be performed. NRC approval is documented in NEDE–24011–P–A–14–US, May 2000.