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Fuel Transition Report for Hope Creek Generating Station

GE Energy, Nuclear

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Fuel Transition Report
For
Hope Creek Generating Station

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ACRONYMS AND ABBREVIATIONS

Term	Definition
AOO	Anticipated Operational Occurrence
AP	Annulus Pressurization
APF	Axial Peaking Factor
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARTS	APRM, RBM, and Technical Specifications Improvement Program
ATWS	Anticipated Transient Without Scram
BPWS	Banked Position Withdrawal Sequence
BSP	Backup Stability Protection
Btu	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CACS	Containment Atmosphere Control System
CLTP	Current Licensed Thermal Power
CPR	Critical Power Ratio (ΔCPR = Change in CPR)
CRGT	Control Rod Guide Tube
°F	Degrees Fahrenheit
DIVOM	Delta CPR over Initial MCPR Versus the Oscillation Magnitude
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Day
ELLLA	Extended Load Line Limit Analysis
EOC	End-of-Cycle
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guideline
FHA	Fuel Handling Accident
FPCC	Fuel Pool Cooling and Cleanup System
ft	Foot
GE	General Electric Company
GNF	Global Nuclear Fuel - LLC
GWd/MT	Gigawatt-Days per Metric Ton
HCGS	Hope Creek Generating Station
HDSFSR	High Density Spent Fuel Storage Rack
HDFS	High Density Fuel Storage

Term	Definition
hr	Hour
ICA	Interim Corrective Action
ICF	Increased Core Flow
in^2	Square inches
in^4	Inches to the 4^{th} power
JR	Jet Reaction
k_{eff}	Effective Multiplication Factor
k_∞	Infinite Multiplication Factor
kW	Kilowatt
lbf	Pounds-force
lbm	Pounds-mass
lbs	Pounds
LDFS	Low Density Fuel Storage
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPF	Local Peaking Factor
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MSIV	Main Steam Isolation Valve
MWt	Megawatts thermal
NFV	New Fuel Vault
NFWT	Normal Feedwater Temperature
NRC	Nuclear Regulatory Commission
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OPRM	Oscillation Power Range Monitor
PBDA	Period Based Detection Algorithm
PCT	Peak Cladding Temperature
ppm	Parts per million
PSEG	PSEG Nuclear LLC
psia	Pounds per square inch, absolute
psid	Pounds per square inch, difference
RBM	Rod-Block Monitor

Term	Definition
RCIC	Reactor Core Isolation Cooling
RFWT	Reduced Feedwater Temperature
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Differences
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RSLB	Recirculation Suction Line Break
RTP	Rated Thermal Power
SER	Safety Evaluation Report
SFPC	Spent Fuel Pool Cooling
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-Loop Operation
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake
TAF	Top of Active Fuel
TLO	Two-Loop Operation

ABSTRACT

The implementation of a new fuel design for a General Electric (GE) Boiling Water Reactor (BWR) follows a two-step process. First, the new fuel design is submitted to and approved by the Nuclear Regulatory Commission (NRC) [[ ]] via the GESTAR II Amendment 22 process. Then, plant-specific analyses are performed to justify use of the new fuel design in an upcoming plant reload. The [[ ]] analyses consist of one-time [[ ]] analyses and [[ ]] analyses. The [[ ]] analyses have been performed to support introduction of the GE14 fuel design at Hope Creek Generating Station (HCGS) for the Current Licensed Thermal Power of 3339 MWt. The [[ ]] analyses are performed for each reload regardless of fuel design.

This report summarizes the results of the [[ ]] analyses and evaluations for the Hope Creek Generating Station (HCGS) mixed cores of GE14 and SVEA-96+ fuel up to and including a full core of GE14. The mixed fuel cores will consist of approximately 20% GE14 and 80% SVEA-96+ for the initial cycle, 50% GE14 / 50% SVEA-96+ and 80% GE14 / 20% SVEA-96+ for the next two cycles, respectively, until essentially the entire core will contain GE14 in the following cycle. The major [[ ]] analyses are the stability evaluation, decay heat assessment, Reactor Internal Pressure Differences (RIPD) and structural assessment, Reactor Recirculation Pump Seizure Event during Single-Loop Operation (SLO), and Appendix R evaluation. In addition, other technical issues, such as the plant response to an Anticipated Transient Without Scram (ATWS), a Fuel Handling Accident (FHA), the seismic response, a neutron fluence evaluation, Banked Position Withdrawal Sequence (BPWS) acceptability, radiation source term, Emergency Procedure Guidelines applicability verification, fuel storage criticality requirements, mechanical compatibility, hydrogen generation / recombination analysis, and Standby Liquid Control System (SLCS) margin criteria for SVEA-96+ fuel are presented. The results of the [[ ]] analyses and evaluations conclude that HCGS can safely load and operate using GE14 fuel.

1.0 INTRODUCTION AND SUMMARY

The implementation of a new fuel design for a General Electric (GE) Boiling Water Reactor (BWR) follows a two-step process. First, the new fuel design is submitted to and approved by the Nuclear Regulatory Commission (NRC) [[ ]] via the GESTAR II Amendment 22 process. Then, plant-specific analyses are performed to justify use of the new fuel design in an upcoming plant reload. The [[ ]] analyses consist of one-time [[ ]] analyses and [[ ]] analyses. The [[ ]] analyses have been performed to support introduction of the GE14 fuel design at Hope Creek Generating Station (HCGS) for the Current Licensed Thermal Power of 3339 MWt. The [[ ]] analyses are performed for each reload regardless of fuel design.

HCGS will be loading GE14 fuel for Cycle 13 operation. Currently, the plant is operating with non-GE14 fuel assemblies (SVEA-96+) in the core. [[ ]] analyses will be performed and documented in the plant and cycle-unique Supplemental Reload Licensing Report.

This report summarizes the results of the [[ ]] analyses and evaluations for the HCGS mixed cores of GE14 and SVEA-96+ fuel up to and including a full core of GE14. The mixed fuel cores will consist of approximately 20% GE14 and 80% SVEA-96+ for the initial cycle, 50% GE14 / 50% SVEA-96+ and 80% GE14 / 20% SVEA-96+ for the next two cycles, respectively, until essentially the entire core will contain GE14 in the following cycle. The following major [[ ]] analyses were performed considering the most limiting operating domain, such as the Extended or Maximum Extended Load Line Limit Analysis (ELLLA/MELLLA) and/or Increased Core Flow (ICF) as applicable:

- Stability Evaluation;
- Decay Heat Assessment;
- Reactor Internal Pressure Differences (RIPD) and Structural Assessment;
- Reactor Recirculation Pump Seizure Event during Single-Loop Operation (SLO); and
- Appendix R Evaluation.

The results of the stability evaluation indicate that the introduction of GE14 fuel will not affect the ability of HCGS to comply with the requirements of GDC 12 (10 CFR 50 Appendix A Criterion 12).

The introduction of GE14 fuel assemblies results in an insignificant change in decay heat. In general, assuming that reactor power level, effective full power days (EFPDs), and enrichment do not change relative to the decay heat bases, the effect of fuel product line on decay heat is well within the overall uncertainty in the calculation.

The results of the RIPD evaluation are used as input to the reactor internals structural integrity evaluation. The RIPD results for the Normal and Upset conditions indicate that the pressure differences across the internal components, except for the fuel channel walls, for the mixed core configuration are bounded by the full core configuration of a single fuel type. The pressure differences across the SVEA-96+ channel walls for the mixed core configuration are only 1% higher than the full core configuration of SVEA-96+ fuel, which is considered to be insignificant

and within the range of calculational uncertainty. The introduction of GE14 fuel has no effect on acoustic and flow-induced loads.

The Structural Assessment was performed consistent with the HCGS design basis load definitions and load combinations. The load changes for the reactor internals, as a result of the GE14 fuel introduction, can be attributed to a change in the RIPDs. The reactor internals remain qualified for GE14 fuel, including a mixed transition core of GE14 and SVEA-96+ fuels.

The Operating Limit Minimum Critical Power Ratio (OLMCPR) due to a postulated Recirculation Pump Seizure during Single Recirculation Loop Operation (SLO) has been determined to be 1.51 in order to protect an assumed Safety Limit Minimum Critical Power Ratio (SLMCPR) of 1.12.

An Appendix R assessment concluded that the maximum PCT remains below the 1500°F limit. Further, the water level response for the full core GE14 bounds the transitional cores of SVEA-96+ and GE14 fuels. Therefore, both the GE14 and the SVEA-96+ fuel integrities for Appendix R are maintained.

Other topics included in this report are:

- Anticipated Transient Without Scram (ATWS)
- Seismic Response
- Neutron Fluence
- Fuel Handling Accident (FHA)
- Radiation Source Term
- Emergency Operating Procedure Data
- Fuel Storage Criticality Requirements
- Mechanical Compatibility
- Hydrogen generation / recombination analysis
- Standby Liquid Control System (SLCS) margin criteria for SVEA-96+ fuel
- Banked Position Withdrawal Sequence (BPWS) Acceptability

The above topics are shown to be acceptable from a GE14 fuel introduction perspective. [[

]]

The results of the [[ ]] analyses and evaluations contained in this report conclude that HCGS can safely load and operate using GE14 fuel including the transition cycles with SVEA-96+ fuel.

1.1 GE14 FUEL BUNDLE DESIGN

A GE14 bundle schematic is shown in Figure 1-1. The GE14 design consists of 92 fuel rods and two large central water rods contained in a 10x10 array. The two water rods encompass eight fuel rod positions. Fourteen of the fuel rods terminate just past the top of the 5th spacer and are designated as part length fuel rods. Eight fuel rods are used as tie rods. The GE14 lattice arrangement is shown in Figure 1-2. The rods are spaced and supported by the upper and lower tie plates and eight spacers over the length of the fuel rods. This assembly is encased in an

interactive fuel channel, which has been used on the GE10, GE11, GE13, and GE12 designs. Finger springs control the coolant leakage flow between the lower tie plate and the channel.

The fuel rods consist of high-density ceramic uranium dioxide or urania-gadolinia fuel pellets stacked within Zircaloy-2 cladding. The cladding will generally have an inner zirconium liner. The fuel rod is evacuated and backfilled with helium.

1.1.1. New Design Features

GE14 was designed for mechanical, nuclear, and thermal-hydraulic compatibility with the other GE fuel designs. In addition, this compatibility can also be demonstrated for the use of GE14 with non-GE fuel designs. The GE14 design includes many features of the GE10, GE11/13 and GE12 fuel designs including optional PCI resistant barrier cladding, high performance spacers, part length rods, interactive thick corner/thin wall channel, and axial Gd and enrichment loading. New or improved features included in the GE14 are:

- Optimized spacer positions and part length rod length
- Eight High Performance Zr-2 spacers
- TriClad™ cladding (optional)
- Debris filter lower tie plate as standard equipment

A discussion of each of these new design features is provided below.

1.1.1.1 Fuel Assembly Configuration

The 10x10 fuel assembly configuration is described above and shown in Figure 1-1 and Figure 1-2. Although the GE14 design operates at a reduced LHGR compared to previous designs (GE11 and GE13), barrier fuel cladding is still utilized to avoid capacity factor losses due to PCIOMR type restrictions that would be required for non-barrier cladding. Therefore, the patented zirconium barrier liner that has been incorporated in GE6 through GE12 designs is again included in the GE14 design. Non-barrier cladding is also offered as an option. Fuel licensing criteria are met for the standard barrier; optional non-barrier cladding and optional TRICLAD™ design. The TRICLAD™ design incorporates an inner zircaloy-2 clad layer on the standard zirconium liner to slow post failure degradation of the cladding.

As with the previous GE11/12/13 design implementation, the combination of number of part length rods and length of part length rods has been optimized to maximize the fuel weight while maintaining the pressure drop and stability characteristics of previous designs. There are 78 full-length fuel rods in the GE14 lattice. The initial fuel rod pre-pressurization is the same as the GE11/12/13. The nominal pellet density is also the same as GE11/12/13.

1.1.1.2 Part Length Rods

Fourteen (14) part length fuel rods (PLRs) are selectively located in the lattice as shown in Figure 1-2 to reduce two-phase pressure drop. These PLRs are approximately 2/3 of the length of the full-length rods and terminate just above the top of the 5th spacer and are one of the key factors that enable this design to match the pressure drop and stability performance of currently operating fuel designs, while maximizing fuel weight. The PLRs increase the moderator-to-fuel ratio in the top of the core in the cold state, which improves cold shutdown margin for the same

lattice enrichment. This improvement will allow higher enrichment and/or a more reactive (hot) core for the same cold shutdown margin.

1.1.1.3 Debris Filter Lower Tie Plate (LTP)

The GE14 LTP is identical to the GE12 LTP. GE14 is assembled with a debris filter lower tie plate as standard equipment. It is an option to use a non-debris filter LTP. The debris filter LTP has an underlying grid, which screens out the debris. Over the many years of nuclear power plant operation, some fuel failures have occurred due to small amounts of debris (wire, springs, drill turnings, etc). Such debris accumulates in the bottom plenum of the reactor vessel, and may be swept into the fuel assembly where it may be lodged in the fuel assembly structure. Once lodged in the fuel assembly structure, flow induced vibration of the debris can cause fretting wear on the rods that may eventually lead to a failure of a fuel rod. The debris filter LTP mitigates debris related fuel rod failures by reducing the size of debris that can enter the fuel assembly.

The primary approach to achieving good channel hydraulic stability is to minimize the ratio of two-phase to single phase bundle pressure drop in the fuel assembly. As with prior designs both the debris filter and non-debris filter LTP provide single phase pressure drop which, when combined with the two phase pressure drop reduction from use of the PLRs, low pressure drop spacers, and upper tie plate, maintains adequate channel stability.

Both LTP designs include the GE feature of an extended boss around each of the water rod lower end plugs. These extended bosses preclude the occurrence of unacceptable flow induced vibrations that could otherwise be caused by coolant impinging directly on the longer water rod lower end plugs.

1.1.1.4 High Performance Spacers

The GE14 bundle uses eight GE12 Zircaloy ferrule spacers with Alloy X-750 springs. The GE12 Zircaloy ferrule spacer is similar to the ferrule spacer designs used in GE9, 10, 11 and 13. The 14 ferrules in the PLR lattice positions have been removed from the three spacers above the PLRs to minimize the two–phase pressure drop.

The GE14 bundle design uses non–uniform axial spacing of the eight spacers to optimize critical power performance. The GE14 spacer positions result in an improvement in the CPR, and a lower two-phase pressure drop that maintains stability performance compared to previous GE designs.

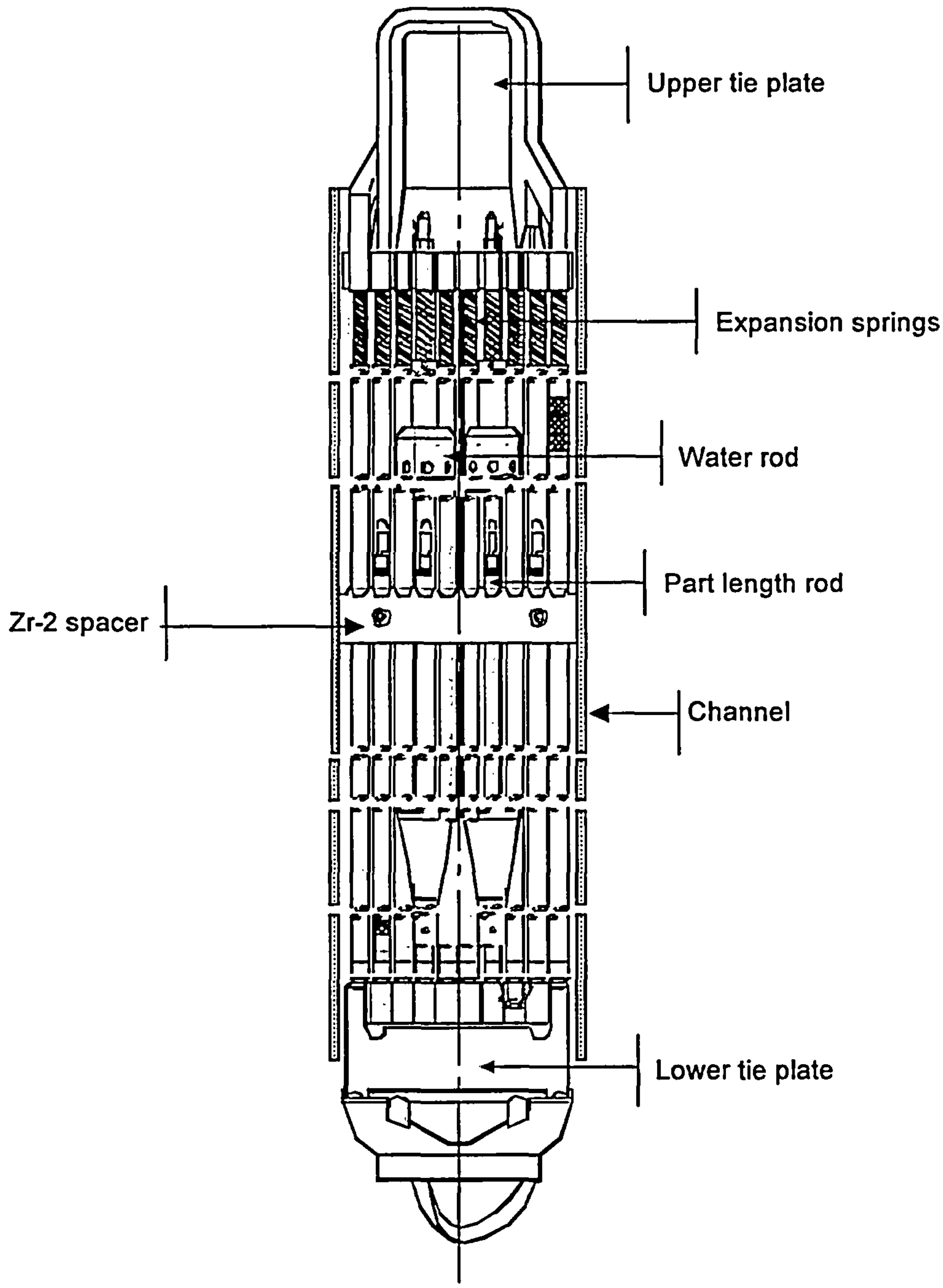
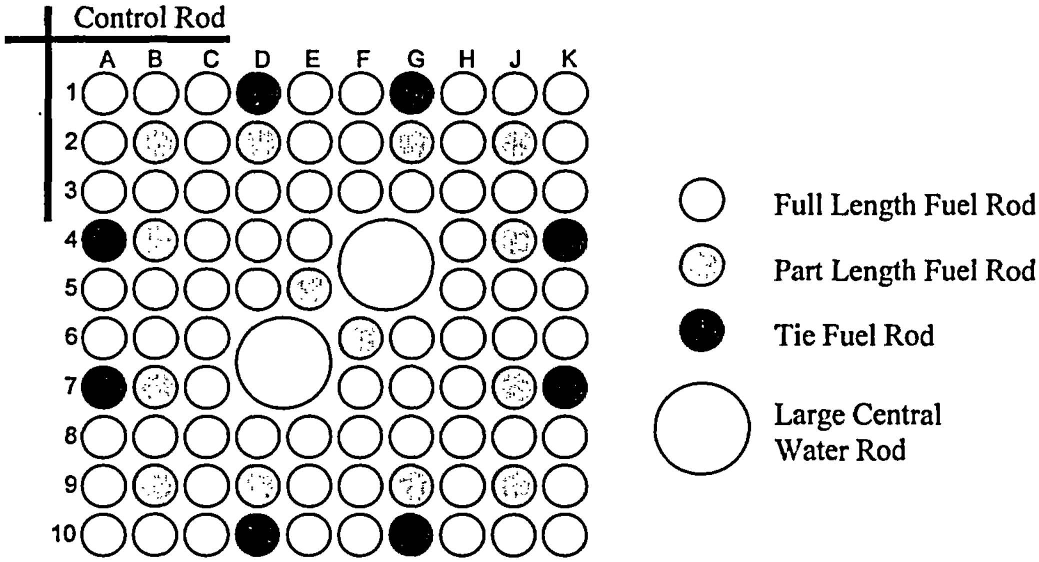


Figure 1-1
GE14 Bundle Assembly



Notes: 1) View of bundle lattice looking down from top.

2) Channel fastener is at A1 corner.

Figure 1-2
GE14 Lattice Arrangement

2.0 STABILITY EVALUATION

2.1 INTRODUCTION

HCGS is introducing GE14 new fuel to replace discharged fuel for the Cycle 13 core design. HCGS will also implement stability solution Option III (Reference 3) beginning in Cycle 13. The objective of this section is to document the effect of the introduction of GE14 fuel on Stability for HCGS. The GE14 new fuel introduction complies with the current licensing requirements for stability Option III. Product line dependence on stability margins is addressed separately in Section 2.9 of the Amendment 22 compliance document (Reference 2).

2.2 OPTION III EVALUATION

Option III is a detect and suppress solution which combines closely spaced Local Power Range Monitor (LPRM) detectors into "cells" to effectively detect either core-wide or regional modes of reactor instability. These cells are termed oscillation power range monitor (OPRM) cells and are configured to provide local area coverage with multiple channels. Plants implementing Option III have installed new hardware to combine the LPRM signals and to evaluate the cell signals with instability detection algorithms. Of these algorithms, only the Period Based Detection Algorithm (PBDA) is officially credited in the Option III licensing basis. This algorithm provides an instrument setpoint designed to trip the reactor before an oscillation can increase to the point where the Safety Limit Minimum Critical Power Ratio (SLMCPR) is exceeded.

The current stability reload licensing basis is to calculate the limiting Operating Limit Minimum Critical Power Ratio (OLMCPR) required to protect the SLMCPR for both steady-state and transient conditions as specified in the Option III methodology. The OLMCPR is calculated for a range of OPRM setpoints. Selection of an appropriate instrument setpoint can then be made based upon the actual OLMCPR to provide adequate SLMCPR protection.

The PBDA setpoint calculation requires the use of the regional DIVOM (which is defined as the Delta CPR over Initial MCPR Versus the Oscillation Magnitude) curve. Past TRACG evaluations by GE have shown that the generic DIVOM curves specified in NEDO-32465-A (Reference 3), might not be conservative for current plant operating conditions for plants that have implemented Stability Option III. Specifically, a non-conservative deficiency has been identified for high peak bundle power-to-flow ratios in the generic regional mode DIVOM curve. The deficiency results in a non-conservative slope of the associated DIVOM curve so that the Option III trip setpoint may be too high. GE has made a Part 21 Notification (Reference 4) on this issue.

The DIVOM portion of the Option III methodology has the potential to be affected by the introduction of the GE14 fuel product line. This is recognized in GESTAR-II and is the basis for allowing the use of an alternate appropriate DIVOM slope. Because the procedure specifically provides a suitable DIVOM slope on a cycle specific basis, as allowed by the GESTAR-II licensing basis, the Option III solution is fully capable of supporting GE14 fuel introduction.

The BWR Owners Group (BWROG) Stability-II committee has addressed the DIVOM issue by providing guidance on a DIVOM generation procedure (Reference 5) to resolve the Part 21 issue.

One of the criteria used by this committee is to verify that any long-term solution is compatible with current fuel designs; therefore, GE14 applicability will be part of that work. Furthermore, any changes to the current licensing methodology will require review and approval by the NRC before they can be implemented.

In addition, prior to arming the Option III system, PSEG will address the Part 21 issues and Safety Communications (References 5-8). Among these concerns are:

1) Effect of signal attenuation from conditioning filter on the OPRM setpoints,
2) Recommended PBDA settings for the Option III, and
3) Adequacy assessment of the current OPRM Enabled region.

2.3 ICA EVALUATION

Should the Option III system be declared inoperable, stability Interim Corrective Actions (ICAs, Reference 9) are incorporated as alternate methods/procedures and restrict plant operation in the high power, low core flow region of the BWR power/flow-operating map. The ICAs contain specific operator actions to provide clear instructions (depending upon plant type) for operator response to a reactor inadvertently (or under controlled conditions) entering any of the defined restricted regions. ICAs provide appropriate guidance to reduce the likelihood of instability and to enhance early detection in the very unlikely event that some stability threshold is exceeded in spite of the ICA guidelines. Because the ICAs are a generic interim solution that is approved to cover all operations and accident scenarios, the GE14 new fuel introduction is covered under this generic licensing basis assuming that the operators recognize the onset of power oscillations and take action to scram the reactor when thermal-hydraulic oscillations are observed as dictated in Reference 9.

PSEG has committed to review the applicability of the ICA regions on a cycle-specific basis, and take appropriate action to revise the ICA regions if needed. The effectiveness of the ICA regions in preventing reactor instability will be evaluated using the NRC-approved PANACEA/ODYSY methodology (Reference 10) against the NRC approved ODYSY stability acceptance criterion for each subsequent reload cycle, including the HCGS Cycle 13 with GE14 new fuel introduction.

PSEG will use the Backup Stability Protection (BSP) (Reference 11) evaluation to validate or expand the ICA regions to ensure adequacy of stability protection.

2.4 CONCLUSION

The GE14 new fuel introduction complies with the current licensing requirements for stability Option III. Product line dependence on stability margins is addressed separately in Section 2.9 of the Amendment 22 compliance document (Reference 2). The Option III solution is fully capable of supporting GE14 fuel introduction, because the actual core designs are used to produce a suitable DIVOM slope, as allowed by GESTAR-II licensing basis. Should the Option

III system be declared inoperable, the ICA regions effectiveness evaluation is fully capable of supporting GE14 fuel introduction, because this evaluation will be performed on a cycle specific basis.

3.0 DECAY HEAT

3.1 INTRODUCTION

A comparative assessment was made to determine the effect of the introduction of GE14 fuel on the current decay heat basis for HCGS.

The analyses and evaluations support operation of HCGS for the following conditions:

- Transitional cores of SVEA-96+ and GE14 fuel
- Full cores of GE14 fuel
- Current Licensed Thermal Power
- Extended Load Line Limit Analysis (ELLLA) and MELLLA power flow operating domains

3.2 CORE DECAY HEAT ANALYSIS

This analysis covers HCGS transition cycles from SVEA-96+ to GE14 as well as an entire core of GE14. This analysis provides justification for the continued use of the current HCGS decay heat bases by establishing that the decay heat bases are not adversely affected by the fuel design changes and that the current bases are conservative when applied to SVEA-96+ and GE14.

In general, decay heat is not a function of fuel product line or fuel manufacturer. This was proven for SVEA-96+ by performing a decay heat calculation using the ANS 5.1-1979 Standard for a single specific SVEA-96+ fuel bundle and then comparing that result to a similar calculation for a GE14 fuel bundle of comparable enrichment. The results are within 0.2% of each other for over 10^8 seconds and are always within 1%. The magnitude of this difference is within the calculation uncertainty and does not represent a different decay heat response between SVEA-96+ and GE14. This result provides justification that the current HCGS decay heat bases remain valid with the introduction of GE14 fuel.

A further justification for the continued use of the May-Witt decay heat table as the design and licensing basis for HCGS is provided by its conservative nature. The May-Witt table is a conservative estimate of decay heat that bounds predictions from currently accepted standards such as ANS 5.1-1979. Consequently, the current HCGS May-Witt design and licensing basis will continue to be applicable to all transition cycles and the full core GE14 cycles. A comparison of May-Witt with a conservative application of ANS 5.1-1979 for HCGS is shown in Figure 3-1[1]. Therefore, the introduction of GE14 fuel at HCGS will not invalidate the current core decay heat bases.

3.3 SPENT FUEL POOL COOLING DECAY HEAT ANALYSIS

The Spent Fuel Pool Cooling (SFPC) analysis for HCGS consists primarily of a determination of the design heat load (after a normal batch discharge) and the maximum anticipated heat load (after a full core discharge). Additional analyses include alternate shutdown cooling with natural

1 Because the HCGS May-Witt bases includes fuel relaxation energy and the ANS 5.1 bases does not, the comparison starts at 120 seconds, which is beyond the maximum time for fuel relaxation energy.

circulation and fuel pool cooling with Residual Heat Removal (RHR) Fuel Pool Cooling and Cleanup System (FPCC) assist mode.

The current decay heat basis for SFPC is Auxiliary Systems Branch (ASB) 9-2. The ASB 9-2 table does not depend on fuel parameters such as enrichment and therefore applies to both GE14 and SVEA-96+.

The SFPC analyses are generally based on a conservative set of assumptions about the loading of the pool. Typically the discharge batch from an equilibrium cycle is used to completely fill the pool, while leaving enough room for a full core discharge. Although the discharge batch size may be slightly different for GE14 relative to SVEA-96+, the difference in the batch size is not large enough to affect the conservatism in the SFPC analysis.

The Alternate Shutdown Cooling with Natural Circulation analysis for HCGS is based on ANSI/ANS 5.1-1979 with a two-sigma allowance for uncertainty. The supplemental analysis for coastdown power levels is a best estimate calculation, based on ANSI/ANS 5.1-1994 with no allowance for uncertainty. In both cases, the change in fuel design from SVEA-96+ to GE14 does not change the decay heat evaluation.

The decay heat basis for the HCGS fuel pool-cooling analyses that utilize RHR in a FPCC assist mode is ASB 9.2. As stated earlier, this table applies equally well to GE14 and SVEA-96+.

3.4 CONCLUSIONS

The transition from SVEA-96+ to GE14 is acceptable relative to analyses based on May-Witt. For other decay heat bases, the decay heat for the two fuel designs is shown to be the same.

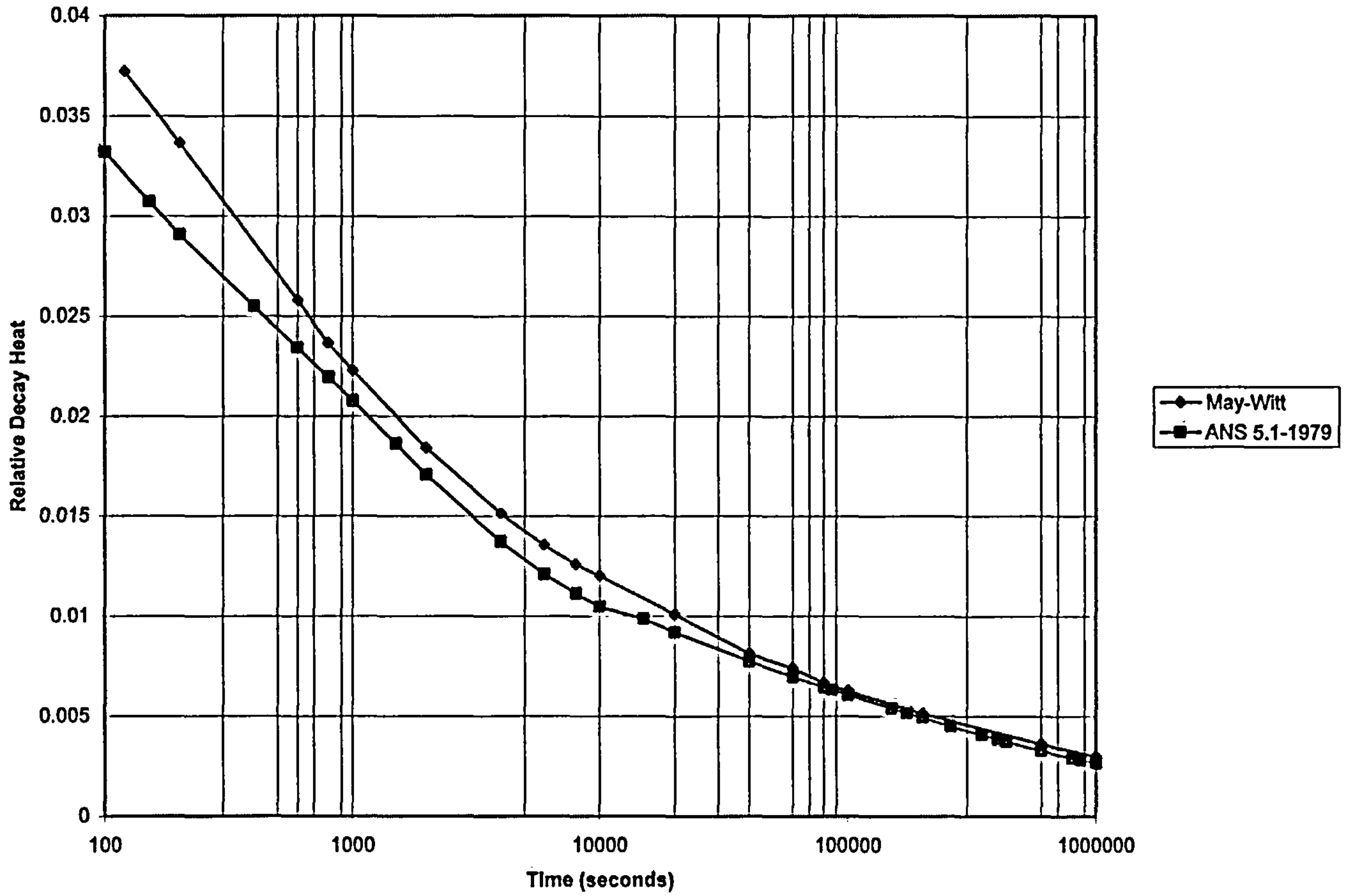


Figure 3-1
Comparison Between The Current Design Basis (May-Witt) And A Conservative Application Of ANS 5.1-1979

4.0 REACTOR INTERNAL PRESSURE DIFFERENCES

Operation with a mechanically different fuel design can affect the pressure differences across reactor internal components because the core hydraulic resistance can change. Operation in the ICF domain results in higher initial core flow relative to rated core flow conditions and therefore yields higher pressure difference across the components. ICF bounds operation at lower core flows such as those for the ELLLA / MELLLA domain. The lower limit of rated feedwater temperature (i.e., reduced feedwater temperature (RFWT)) of -22°F is also considered in the analysis because the lower steam generation in the core can increase the depressurization rate. The analysis was performed at two different power levels, current licensed thermal power (CLTP) equal to 3339 MWt and 120% of the original licensed thermal power (OLTP) equal to 3952 MWt. The reactor internal pressure differences (RIPD) results are used as inputs to the structural evaluation to determine the reactor internal structural integrity.

The fuel lift margin and control rod guide tube (CRGT) lift force are also analyzed for the limiting ICF conditions at both power levels. The fuel lift margin and CRGT lift force at ELLLA / MELLLA conditions are also bounded by ICF as a result of the higher core pressure drops.

Therefore, operation in the ICF domain with consideration of normal feedwater temperature (NFWT) and RFWT at both power levels is the basis for the following discussion.

4.1 ANALYSIS APPROACH AND INPUTS

Analyses of Normal operating conditions were performed with the steady-state thermal-hydraulic model ISCOR[1] at CLTP and 120% OLTP Rated Thermal Power (RTP) with 105% of rated core flow. The inputs used for this analysis are consistent with the assumption of both GE14 and the SVEA-96+ fuel designs for pressure drop consideration. The analysis considers two core configurations: a full core of a single fuel type, and a mixed core of GE14 and SVEA-96+ fuel. Conservative adders and multipliers were [[ ]] applied to the steady-state Normal condition pressure differences to calculate the Upset RIPDs based on GE technical design procedures.

The Emergency and Faulted RIPD values were calculated using the LAMB model[2] to analyze the limiting inadvertent actuation of automatic depressurization system valves, and main steam line break inside containment accident, respectively. The steam dryer pressure drop is determined for main steam line break outside containment. Both Emergency and Faulted conditions are analyzed at high power conditions, (102% multiplier was applied to both CLTP and 120% OLTP per Regulatory Guide 1.49) and 105% of rated core flow. The Faulted condition RIPD calculation also includes an evaluation at the low power cavitation interlock

[1] The ISCOR code is not approved by the NRC by its name for RIPD. However, the Safety Evaluation Report (SER) supporting approval of GE report NEDE-24011P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in reactor internal pressure differences application is consistent with the approved models and methods.
[2] The LAMB code is approved for use in Emergency Core Cooling System (ECCS) - LOCA applications (GE reports NEDE-20566P-A and NEDO-20566A), but no approving SER exists for the use of LAMB in the evaluation of reactor internal pressure differences. The use of LAMB for these applications is consistent with the model description of NEDE-20566P-A.

point. The lowest power and highest core flow condition (interlock point) is approximately 790 MWt and 105% of rated core flow. The analyzed interlock power/flow point is conservative due to higher mismatch between the steam generation in the core and the steam leaving the reactor vessel. Similarly, the higher steam generation rate to break flow rate mismatch due to the RFWT condition is also evaluated at both the high power and interlock conditions.

4.2 RIPD ANALYSIS RESULTS

The results of the RIPD analyses for a full core configuration of single fuel type are shown in Table 4-1 through Table 4-6. The RIPD results for the Normal and Upset conditions indicate that the pressure differences across the internal components, except for channel walls, for the mixed core configuration are bounded by the full core configuration of single fuel type. The pressure differences across the SVEA-96+ channel walls for the mixed core configuration are only 1% higher than the full core configuration of SVEA-96+ fuel, which is considered to be insignificant and within the range of calculation uncertainty. The LAMB code used to calculate the Emergency and Faulted RIPD values is only capable of analyzing a full core of a single fuel type. Therefore, the final Emergency and Faulted RIPD results presented here are calculated by applying an additional [[ ]] conservatism to the LAMB output values based on GE technical design procedures. This [[ ]] conservatism multiplier offsets the small difference in the SVEA-96+ channel wall pressure differences due to core configurations (full core vs. mixed core). Therefore, the RIPD results for the full core of single fuel type are reported, which is consistent with the GE RIPD methodologies.

The introduction of GE14 fuel has no effect on the acoustic and flow-induced loads, which are caused by pressure waves as a result of a recirculation suction line break. The magnitudes of these loads are not dependent on the fuel type; they are dependent upon the initial pressures and temperatures of the fluid in the downcomer region (outside the core shroud) and the geometry of the vessel internals, the core shroud and jet pumps, which remain unchanged for new fuel design.

Table 4-1
HCGS Normal and Upset Conditions Reactor Internal Pressure Differences

Fuel Type:	GE14	GE14	SVEA-96+	SVEA-96+	GE14	GE14
Condition:	3339 MWt NFWT Normal	3406 MWt NFWT Upset	3952 MWt NFWT Normal	4031 MWt NFWT Upset	3952 MWt NFWT Normal	4031 MWt NFWT Upset
Reactor Internal Components[1]	(psid)	(psid)	(psid)	(psid)	(psid)	(psid)
Shroud Support Ring and Lower Shroud	24.44	26.84	26.81	29.21	27.09	29.49
Core Plate and Guide Tube	18.40	20.80	19.43	21.83	19.70	22.10
Upper Shroud	6.25	9.38	7.59	11.38	7.59	11.39
Shroud Head	7.13	10.70	8.52	12.78	8.52	12.78
Shroud Head to Water Level, Irreversible	9.68	14.51	11.22	16.83	11.22	16.83
Shroud Head to Water Level, Elevation	0.86	1.29	0.75	1.12	0.75	1.12
Core Average Power Bundle	8.41	11.31	10.74	13.64	9.69	12.59
Central Average Power Bundle	9.26	12.16	11.76	14.66	10.63	13.53
Maximum Power Bundle	10.62	13.52	13.23	16.13	12.11	15.01
Top Guide	0.54	0.58	0.54	0.58	0.54	0.58
Steam Dryer	0.34	0.44	0.47	0.61	0.47	0.61

1. The normal and upset conditions RIPDs for the RFWT condition are bounded by the NFWT condition.

Table 4-2
HCGS Emergency Condition Reactor Internal Pressure Differences

Fuel Type:	SVEA-96+	GE14	SVEA-96+	GE14
Condition:	High Power 4031 MWt[1] NFWT	High Power 4031 MWt[1] NFWT	High Power 4031 MWt[1] RFWT	High Power 4031 MWt[1] RFWT
Reactor Internal Components[2]	(psid)	(psid)	(psid)	(psid)
Shroud Support Ring and Lower Shroud	32.0	33.0	31.0	33.0
Core Plate and Guide Tube	23.0	23.0	21.5	22.0
Upper Shroud	13.6	13.6	12.6	13.2
Shroud Head	13.8	14.0	12.8	13.4
Shroud Head to Water Level, Irreversible	15.6	15.8	14.7	15.2
Shroud Head to Water Level, Elevation	1.4	1.5	1.4	1.5
Core Average Power Bundle	13.1	10.5	11.5	10.8
Maximum Power Bundle	15.7	13.4	14.4	13.5
Top Guide	0.33	0.40	0.29	0.40
Steam Dryer	Note 3	Note 3	Note 3	Note 3

1. Regulatory Guide 1.49, 102% applied.
2. The emergency condition at CLTP power level is bounded by the emergency condition at 120% OLTP due to lower initial power level.
3. Bounded by Faulted conditions due to slower depressurization.

Table 4-3
HCGS Faulted Condition Reactor Internal Pressure Differences

Item	Fuel Type:	GE14	SVEA-96+	GE14	SVEA-96+	GE14
	Condition:	High Power 3430 MWt[1] NFWT	High Power 4031 MWt[2] NFWT/RFWT	High Power 4031 MWt[2] NFWT/RFWT	Interlock 790 MWt NFWT/RFWT	Interlock 790 MWt NFWT/RFWT
	Reactor Internal Components	(psid)	(psid)	(psid)	(psid)	(psid)
1	Shroud Support Ring and Lower Shroud	44.0	44.0 / 43.0	44.0 / 43.0	41.0 / 41.0	43.0 / 43.0
2	Core Plate and Guide Tube	22.5	22.5 / 23.0	23.5 / 24.0	22.0 / 22.0	24.0 / 24.0
3	Upper Shroud	25.5	24.5 / 24.5	24.5 / 24.0	27.0 / 27.0	27.0 / 27.0
4	Shroud Head	26.5	25.5 / 25.0	25.0 / 25.0	27.0 / 27.0	27.5 / 27.5
5	Shroud Head to Water Level, Irreversible	28.5	27.5 / 27.0	27.5 / 27.0	28.0 / 28.0	28.5 / 28.5
6	Shroud Head to Water Level, Elevation	1.4	1.2 / 1.2	1.2 / 1.2	2.4 / 2.5	2.5 / 2.5
7	Core Average Power Bundle (Bottom)	11.5	14.2 / 15.2	12.6 / 13.4	9.0 / 9.0	8.9 / 8.9
8	Maximum Power Bundle (Bottom)	13.3	15.9 / 17.0	14.3 / 15.3	10.3 / 10.2	10.0 / 10.0
9	Top Guide	0.70	0.55 / 0.36	0.56 / 0.36	0.98 / 0.97	1.0 / 1.0
10	Steam Dryer	< 11.0	< 11.0	< 11.0	< 11.0	< 11.0[3]

1. An additional 2.7% power is added to for this case to satisfy Regulatory Guide 1.49 requirements.
2. Regulatory Guide 1.49, 102% applied.
3. Bounded by the design pressure differential of 11 psid at hot standby conditions.

Table 4-4
HCGS Normal and Upset Conditions Fuel Lift Margin and CRGT Lift Force

Item		Fuel Type:	GE14	GE14	SVEA-96+	SVEA-96+	GE14	GE14
		Condition:	3339 MWt NFWT Normal	3406 MWt NFWT Upset	3952 MWt NFWT Normal	4031 MWt NFWT Upset	3952 MWt NFWT Normal	4031 MWt NFWT Upset
	Components		(lbf)	(lbf)	(lbf)	(lbf)	(lbf)	(lbf)
1.	Fuel Lift Margin	Hot Power Bundle	410.5	379.2	365.8	334.5	391.1	359.8
2.		Average Power Bundle	425.2	394.0	377.4	346.1	406.4	375.1
3.	CRGT Lift Force	Average Power Bundle	626.8	692.6	573.7	639.5	658.6	724.4

Table 4-5
HCGS Emergency Condition Fuel Lift Margin and CRGT Lift Force

Item	Fuel Type:		SVEA-96+ [1]	GE14 [1]	SVEA-96+	GE14
	Condition:		High Power 4031 MWt [2] NFWT	High Power 4031 MWt [2] NFWT	High Power 4031 MWt [2] RFWT	High Power 4031 MWt [2] RFWT
	Components		(lbf)	(lbf)	(lbf)	(lbf)
1.	Fuel Lift Margin	Hot Power Bundle	296.4	338.0	313.6	341.4
2.		Average Power Bundle	329.9	352.2	344.5	355.3
3.	CRGT Lift Force	Hot Power Bundle	645.6	731.9	Note 3	Note 3
4.		Average Power Bundle	808.9	890.1	Note 3	Note 3

1. The fuel lift margin and CRGT lift force at CLTP conditions are bounded by the 120% OLTP condition.
2. Regulatory Guide 1.49, 102% applied.
3. Bounded by NFWT.

Table 4-6
HCGS Faulted Condition Fuel Lift Margin and CRGT Lift Force

Item	Fuel Type:		GE14	SVEA-96+	GE14	SVEA-96+	GE14
	Condition:		High Power NFWT 3430 MWt[1]	High Power NFWT/RFWT 4031 MWt[2]	High Power NFWT/RFWT 4031 MWt[2]	Interlock NFWT/RFWT 790 MWt	Interlock NFWT/RFWT 790 MWt
	Components		(lbf)	(lbf)	(lbf)	(lbf)	(lbf)
1	Fuel Lift Margin	Hot Power Bundle	310.8	263.8 / 276.2	298.1 / 309.1	328.3 / 329.7	356.8 / 358.2
2		Average Power Bundle	318.1	287.1 / 299.6	306.6 / 320.4	351.4 / 354.8	353.3 / 354.8
3	CRGT Lift Force	Hot Power Bundle	< 765.4	626.5 / 840.2	Note 3 / 765.4	841.6 / 838.1	1008.6/ Note 4
4		Average Power Bundle	< 931.4	782.0 / 907.7	Note 3 / 931.4	909.1 / 905.3	1070.1/ Note 4

1. An additional 2.7% power is added to satisfy Regulatory Guide 1.49 requirements.
2. Regulatory Guide 1.49, 102% applied.
3. Bounded by RFWT.
4. Bounded by NFWT.

5.0 STRUCTURAL ASSESSMENT

The structural integrity of the reactor internals was assessed for load changes associated with a full core of GE14 fuel and with a mixed core of GE14 and SVEA-96+ fuels. The assessment was performed consistent with the design basis load definitions and load combinations. The effects of GE14 and SVEA-96+ fuels on the Seismic loads, Annulus Pressurization and Jet Reaction (AP and JR) loads, and Fuel Lift loads were evaluated and were found to be insignificant. The Acoustic and Flow Induced loads due to the Recirculation Suction Line Break (RSLB) are not affected by changes in the core loading. The effect of the full core of GE14 and the mixed GE14 and SVEA-96+ core on other loads such as, system flow loads, core flow loads, and temperature effect (thermal) loads, as these affect the structural integrity of the reactor internal components, was also determined to be insignificant.

The loads associated with the reactor internal pressure differences (RIPD) change with the introduction of a new fuel design. The effect of such a change for a mixed core of GE14 and SVEA-96+ and for a full core of GE14 was shown to be acceptable relative to the structural integrity of the reactor internal components.

The structural integrity of the following reactor internals was assessed for the loads associated with a full core of GE14 fuel and with a mixed core of GE14 and SVEA-96+ fuels. The original configurations of the internal components were considered unless a component had undergone permanent structural modification, in which case, the modified configuration was used as the basis for the assessment.

- Shroud
- Shroud Support
- Shroud Head and Steam Separator Assembly
- Fuel Channel
- Core plate
- Top Guide
- Control Rod Drive Housing
- CRGT
- Orificed Fuel Support
- Steam Dryer
- Feedwater Sparger
- Jet Pump
- Core Spray Line and Sparger
- Access Hole Cover
- Core Differential Pressure and Liquid Control Line
- Low Pressure Coolant Injection (LPCI) Coupling

The RIPD loads, Seismic loads, AP and JR loads, fuel lift loads, acoustic and flow induced loads, system flow loads, core flow loads, and thermal loads, as applicable were considered in the assessment of the reactor internals for the Normal, Upset, Emergency, and Faulted conditions. The assessment concluded that the stresses in the reactor internals remain within the original design basis acceptance criteria.

Based on the above structural integrity assessment, the reactor internals remain qualified for GE14, including a mixed core of GE14 and SVEA-96+ fuels.

6.0 REACTOR RECIRCULATION PUMP SEIZURE EVENT

The reactor recirculation pump seizure event is analyzed for Single Loop Operation (SLO) at HCGS. This event is classified as an accident; however, a more conservative acceptance criterion, maintaining the Minimum Critical Power Ratio (MCPR) above the SLMCPR, is used as a limit for this event. The anticipated operational occurrence (AOO) analytical approach of assigning OLMCPR limits to the event is a precautionary measure to ensure avoidance of localized dryout within the bundle.

6.1 ANALYSIS APPROACH AND INPUTS

The reactor recirculation pump seizure is analyzed assuming that only one reactor recirculation loop is operational. The ODYN computer code, documented in Reference 13, is used to perform the calculation. This analysis was performed for the HCGS Cycle 13 transition cycle with GE14 fuel and SVEA-96+ in the core and transient analysis inputs that are consistent with the Reload 12/Cycle 13 analyses. The analyses were performed at 70.0% power and 60% core flow. This represents the maximum power and flow condition for SLO and is bounding for the pump seizure event.

The implementation of an SLO OLMCPR ensures the fuel does not violate the SLO SLMCPR. This approach maintains fuel cladding integrity during the postulated event.

Sufficient conservatism was included in the analysis such that the results are considered bounding for future cycles. The following assumptions are used in the analyses:

1. Reference SLO Safety limit of 1.12. The SLO OLMCPR can be adjusted based on the current or future cycle specific SLO SLMCPR, by utilizing the procedure in Section 6.3.
2. The maximum SLO power and flow point is used as the bases for the transient analysis. This initial condition not only leads to the worst transient response but also invokes the least restrictive power and flow dependent operating limits.
3. The analysis is performed at BOC because it tends to give a low void coefficient.
4. A conservative 0.85 multiplier is applied to the void coefficient in the ODYN analysis. This multiplier is to bound alternate cycle exposures and other core designs through an equilibrium core of GE14 fuel.

6.2 REACTOR RECIRCULATION PUMP SEIZURE EVENT RESULTS

The results of the analysis are:

1. The SLO OLMCPR of 1.51 is required so that the reference SLO SLMCPR of 1.12 is protected in the event of a seizure of the recirculation pump in the active loop.
2. The GE14 and SVEA-96+ critical power ratio (CPR) response to the accident is shown in Figure 6-1.

6.3 SAFETY LIMIT ADJUSTMENT PROCEDURE

The SLO OLMCPR limit, provided in this section, assumes a SLO SLMCPR of 1.12. If the cycle-specific SLMCPR changes then the SLO OLMCPR may be adjusted by the following factor: (Cycle Specific SLMCPR / 1.12).

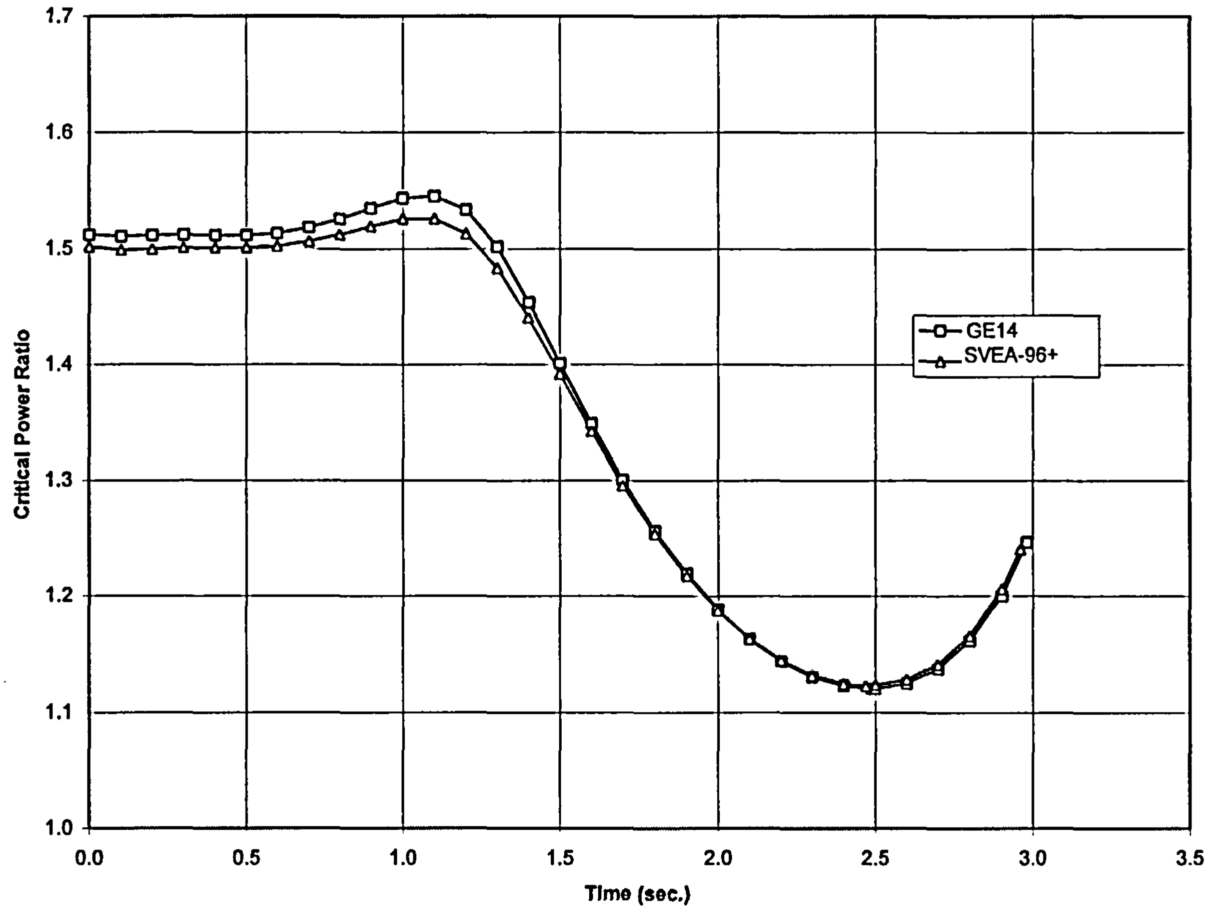


Figure 6-1
GE14 and SVEA-96+ CPR Response to SLO Recirculation Pump Seizure Event

7.0 APPENDIX R

7.1 ANALYSIS APPROACH AND INPUTS

The limiting Appendix R scenario with only reactor core isolation cooling (RCIC) available for vessel inventory makeup at the remote shutdown panel was evaluated using the NRC approved SAFER/GESTR-LOCA and SHEX methodologies (References 14, 15 and 16) assuming that the core configurations consist of GE14 and SVEA96+ fuel types.

The key inputs and assumptions are summarized as follows:

1. The reactor is operating at steady state conditions at full power, normal water level, and 1020 psia dome pressure. For conservatism, the analyzed power is equivalent to 102% of CLTP.
2. The event is initiated by the occurrence of a fire (t = 0 second).
3. Reactor scram occurs at the initiation of the fire.
4. Loss of Offsite Power (LOOP) occurs at the initiation of the fire.
5. The main steam isolation valves (MSIVs) begin to close at event initiation either as a result of the LOOP or due to manual closure. The MSIVs are fully closed in 3.5 seconds.
6. Feedwater flow is assumed to ramp to zero linearly in 5 seconds after the scram.
7. The 1979 ANS 5.1 + 2 sigma decay heat correlation is used for conservatism.
8. For the Remote Shutdown Method, a stuck open safety relief valve (SRV) occurs at event initiation and remains open throughout the event.
9. Operators manually start RCIC from the remote shutdown panel, and injection to the reactor pressure vessel (RPV) begins at 10 minutes into the event. The RCIC pump operates at a constant flow rate of 600 gpm between 150 to 1141 psig.
10. Three SRVs are utilized to depressurize the RPV when reactor water level has been stabilized and suppression pool cooling (SPC) has been initiated. The time to start depressurization will be determined by analysis and is based on meeting the acceptance criteria. The depressurization/cooldown rate is 100 °F/hr. Depressurization may not be needed as the cold RCIC injection water combined with the continuously stuck open SRV may provide adequate depressurization to a RPV pressure where the LPCI system can provide vessel injection.
11. A bounding initial void fraction is used to determine the initial vessel coolant inventory, which is used to assess the water level and Peak Cladding Temperature (PCT) response.
12. SPC starts at 20 minutes and terminates when LPCI injection is initiated.
13. Shutdown cooling (SDC) or alternate shutdown cooling (ASDC) is initiated when the vessel pressure reaches 80 psig and the vessel water level reaches the main steam line (MSL) elevation. Because the LOOP is the limiting event for Appendix R (UFSAR), SDC is not available; therefore ASDC will be initiated instead. The time to initiate ASDC is to be determined by the analysis based on meeting the acceptance criteria.

14. One RHR pump and one RHR heat exchanger is available for both SPC and ASDC modes. The RHR K-factor is 307 Btu/sec-°F with one pump with a flow of 10,000 gpm. The Safety Auxiliaries Cooling System (SACS) temperature is 100°F.

The current licensing basis for Appendix R at HCGS is to maintain the water level above the top of active fuel (i.e., prevent the core from being uncovered).

In the SAFER/GESTR-LOCA model, the core region is represented as the average channel, the bypass, and the hot channel. The bypass region is the volume between the fuel assemblies and the volume between the shroud wall and the peripheral fuel assemblies. The determination of the two-phase water level in the core region is a key parameter in the evaluation of core heat-up during a loss of coolant inventory event. The core water level is generally defined as the mixture water level in the average channel.

Because the inputs and assumptions for calculations with different core configurations are identical, the initial void fraction for both the transition core of GE14 and SVEA-96+ and a full core of a single fuel type of GE14 or SVEA-96+ is the governing parameter in determining the initial vessel inventory affecting the water level and PCT response. Therefore, the bounding initial void fraction is used to assess the water level and PCT response.

7.2 ANALYSIS RESULTS

The results of the Appendix R analyses are shown in Table 7-1 and Table 7-2. As shown in Table 7-1, the water levels in the core bypass region and in the average and hot channels for the limiting case with GE14 are above the top active fuel (TAF). Thus, the core is covered. A full core of GE14 fuel at 102% CLTP yields the bounding void fraction of approximately 48%. Thus, the water level response for the full core GE14 bounds the transitional core of SVEA-96+ and GE14 fuels at CLTP. The calculated PCT is the same as the initial fuel steady state temperatures. Therefore, the fuel integrity is maintained.

The Appendix R analysis was performed at 102% CLTP and 100% rated core flow. For off-rated conditions such as MELLLA, the initial void fraction increase due to a lower core flow (MELLLA) will slightly affect the time to reach the minimum water level. The minimum water levels in bypass and channels never reach the TAF (Table 7-1) and the core remains covered. Thus, the analysis at the rated core flow is applicable in the MELLLA region. In addition, the analysis at the rated core flow bounds ICF as a result of lower initial void fraction due to higher core flow.

The containment analysis results are summarized in Table 7-2. The containment pressures and temperatures are below the respective design limits. Therefore, containment integrity is maintained. These results are independent of fuel design because the decay heat used in the analysis is applicable, or bounding, for both fuel types.

Cold shutdown (vessel temperature less than 200°F) is achieved within 2 hours due to cold RCIC injection and the continuously stuck open SRV. Therefore, the acceptance criterion for reaching cold shutdown within 72 hours is met. This cooldown rate is greater than 100°F/hour and is the result of standard actions to mitigate the event consequences and is acceptable for Appendix R.

Because the stuck open SRV and cold RCIC injection provide adequate depressurization, the suppression pool temperature remains below the heat capacity temperature limit (HCTL) curve. Therefore, the criterion for adequate steam suppression during reactor depressurization is satisfied.

Because the peak suppression pool temperature is less than the current UFSAR value of 212^{0}F, adequate net positive suction head (NPSH) for the ECCS pumps is available. Therefore, the requirement for NPSH is met.

The Appendix R event sequences are summarized in Table 7-3.

The fire barriers, separation of the safe shutdown systems, components, and associated circuits for each fire area/zone of the plant do not change due to the introduction of GE14 fuel. Operation of the plant using GE14 fuel does not affect the fire suppression and detection systems. The physical plant configuration and the combustible loadings are not changed due to the introduction of GE14 fuel.

Table 7-1
Appendix R Analysis Results Summary (RPV)

Parameter	Unit	102% CLTP / 100% F Value	Acceptance Criteria
Minimum Water Level in Bypass Region	ft	>TAF	>TAF or PCT below 1500 °F
Minimum Water Level in average/hot channel	ft	>TAF	>TAF or PCT below 1500 °F
PCT	°F	589[1]	≤1500
Peak Vessel Pressure	psig	1112.1	≤1375
Time to Reach Cold Shutdown Condition (Vessel Temperature below 200 °F)	hr	1.75[2]	≤72

[1] The PCT is the initial fuel steady-state temperature.
[2] The cooldown is greater than 100°F/hr.

Table 7-2
Appendix R Analysis Results Summary (Containment)

Parameter	Unit	102% CLTP / 100% F Value	Acceptance Criteria
Peak Pool Temperature	°F	195.2	≤310
Peak Drywell Pressure	psig	9.1	≤62
Peak Wet well Pressure	psig	9.3	≤62
Peak Drywell Airspace Temperature	°F	300.3	≤340
Peak Wetwell Airspace Temperature	°F	196.2	≤310
Peak Drywell Shell Temperature	°F	226.9	≤310

Table 7-3
Appendix R Event Sequences for Remote Shutdown Method

Event	Time (sec)
Fire Occurs - Loss of Off Site Power - Reactor Scram - One SRV Stuck Open - Control Room Evacuation - MSIVs Begins to Close	0.0
MSIVs Are Closed Fully	3.5
Feedwater Flow Stops	5.0
Operator Starts RCIC Injection (30 Second System Delayed Time Included)	630
Suppression Pool Cooling (SPC) Starts	1200
Vessel Pressure Reaches LPCI Pressure Permissive of 360 psig[1]	1345
Downcomer Minimum Water Level of 29.38 ft above Vessel Zero Is Reached	1751
LPCI Starts Injection* When Maximum Pump Injection Pressure of 286 psig Is Reached and SPC Stops	1864
RCIC Flow Stops When Shutoff Head of 150 psig Is Reached.	2158
Vessel Pressure Reaches Pressure Permissive of 80 psig for ASDC	2415
Water Level Reaches MSL Elevation	2570
ASDC Starts[2]	3600
Cold Shutdown (<200°F) Is Reached	~6300

[1] SORV and cold RCIC injection provide adequate depressurization of the RPV to the vessel pressure where LPCI can inject. Therefore, the action for cooldown at 100°F/hr is not necessary.
[2] ASDC is conservatively assumed to start at 1 hour (3600 seconds) because the conditions for starting ASDC are satisfied at 2570 seconds.

8.0 OTHER TECHNICAL ISSUES

8.1 ATWS

The evaluation of ATWS events is not a design basis requirement. However, it must be demonstrated that the plant is capable of protecting critical components and complying with all applicable licensing criteria during an ATWS event.

ATWS requirements are specified in 10CFR50.62. Boiling Water Reactors (BWR) are required to have an alternate rod insertion system (ARI), automatic recirculation pump trip (RPT), and 86 gpm equivalent boron injection system. These features are included in the Hope Creek plant design. Compliance with these requirements is intended to maintain the integrity of the reactor vessel pressure boundary, the integrity of fuel and a core coolable geometry, and the integrity of the containment. The following criteria are met:

Acceptance Criteria	
Peak Vessel Bottom Pressure	(1500 psig)
Peak Cladding Temperature Cladding Oxidation	(2200°F) (<17%)
Peak Suppression Pool Temperature	(201°F)
Peak Containment Pressure	(62 psig)

The results of a plant-specific ATWS evaluation for HCGS Cycle 13 based on a mixed core of GE14 and SVEA-96+ fuels indicate that deployment of GE14 in the core does not cause adverse effects on the plant ability to meet the ATWS acceptance criteria. This evaluation includes the effects of 1 SRV OOS with the ARTS/MELLLA operating domain. These results are also bounding in the non-ARTS/ELLLA operating domain. The results of the analyses of the limiting ATWS events for Cycle 13 based on the mixed core are shown in the following table.

Event	Exposure	Peak Vessel Pressure (Psig)	Peak Cladding Temperature[1] (°F)	Peak Suppression Pool Temp[2] (°F)	Peak Containment Pressure (Psig)
MSIVC	BOC	1303	1147	169	7.0
MSIVC	EOC	1312	1360	172	7.4
PRFO	BOC	1312	1436	169	7.0
PRFO	EOC	1322	1588	172	7.3

[1] The fuel clad oxidation is insignificant and is less than 17%. Values reported are the limiting of GE14 and SVEA-96+ fuel.

[2] The peak suppression pool temperatures were also validated to meet the ATWS acceptance criteria through depressurization of the reactor. The depressurization evaluation was conservatively performed at 3952 MWt.

For comparison purposes, a HCGS specific ATWS evaluation based on a full core of GE14 fuel is shown in the following table. The full core GE14 fuel ATWS evaluation was performed with parameters and assumptions that were consistent with the Cycle 13 mixed core evaluation, except as noted below.

Event	Exposure	Peak Vessel Pressure (Psig)	Peak Cladding Temperature[1] (°F)	Peak Suppression Pool Temp(°F)	Peak Containment Pressure (Psig)
MSIVC[2]	BOC	1333	<1420	173	7.5
MSIVC[2]	EOC	1334	1420	174	7.8
PRFO[2]	BOC	1343	<1589	173	7.6
PRFO	EOC	1340	1589	176	8.0

1 The fuel clad oxidation is insignificant and is less than 17%. The PCT values under BOC exposure are bounded by end-of-cycle (EOC) exposure for the same event and are not evaluated.
2 A shorter boron transportation delay time of 86 seconds was used in the evaluation of these non-limiting cases. All other cases are evaluated based on an updated boron transportation delay time of 104.4 seconds. The non-limiting cases would be similarly impacted by this revised input and remain bounded by the historically limiting PRFO-EOC case.

8.2 SEISMIC EVALUATION

The HCGS original seismic evaluation was performed using GE6 fuel with 80 mil channels. The maximum horizontal seismic fuel acceleration value for the safe shutdown earthquake (SSE) is 2.02 g and the maximum vertical seismic fuel acceleration value is 0.29 g (Reference 12). The current seismic evaluation was performed to evaluate the effect of the introduction of GE14 fuel. The HCGS core currently contains GE9 and SVEA-96+ fuel, the Cycle 13 core will be a mixed core of GE14 and SVEA-96+ fuel, which is the core that was evaluated for GE14 fuel seismic qualification.

8.2.1. Incore Evaluation

The fuel is a significant mass in the horizontal RPV and internals mathematical model, and is modeled as a separate element. The fuel design and its support result in a first mode of vibration between 4 and 5 Hz. This natural frequency is determined by the mass of the fuel assemblies (primarily the mass of the fuel rods) and the stiffness (primarily the stiffness of the channels). Thus, a change in stiffness of fuel channel results in a change in the frequency of the first fuel mode. As this is in the range of frequencies of high seismic excitation, the effect on the horizontal seismic response is significant. Also, because the fuel is coupled to the shroud, vessel, and shield wall, the seismic response of these structures is also affected.

The horizontal mathematical model used for the HCGS seismic analyses was recreated from the original seismic analyses (Reference 12). This model was benchmarked with the Reference 12 seismic model. Fuel section properties in the horizontal mathematical model were updated for the GE14 and mixed core of GE14 and SVEA-96+ fuel. The following is a comparison of the various fuels sectional properties.

Fuel Type	Cross-Section Area (in^2)	Moment of Inertia (in^4)	Weight (Lbs)
GE6-80	1.657	7.769	666
GE14-100/65/50	1.443	7.004	646
SVEA-96+	1.849	6.761	646

Using these new fuel properties, two eigen analyses were performed for the full core of GE14 and the mixed core of GE14 and SVEA-96+ fuel. The eigen values comparison shows that there is minimal change in the fuel frequency for both cases. The effect of introducing GE14 fuel thus leads to small changes in structural response, which are considered insignificant when compared to the HCGS design bases horizontal and vertical loads.

Therefore, it is concluded that design basis HCGS seismic loads of 2.02 g horizontal and 0.29 g vertical are also valid for GE14 fuel with 100/65/50 channel. Note that there is no change in HCGS AP loads due to the introduction of GE 14 because AP loads are high frequency loads.

8.2.2. New Fuel Vault Seismic/Dynamic Qualification

The original HCGS New Fuel Vault (NFV) seismic qualification is based on linear response spectrum analyses. Also, three independent seismic models represent the NFV assemblies horizontally, representing the top, middle, and bottom portions of the vault assembly. The horizontal fundamental frequencies for the top, middle, and bottom sections are in the range of 14.5 Hz, 11.0 Hz, and 14.8 Hz, respectively.

As discussed in the incore evaluation above, the results of actual HCGS horizontal primary structure eigen analyses, shows that there is minimal change in the fuel frequency for both cases, i.e., GE14 fuel and the mixed core of GE14 and SVEA-96+ fuel. For the HCGS NFV, the less than 3.39% reduction in the horizontal fuel assembly natural frequencies, due to the introduction of GE14 fuel, does not significantly change the NFV seismic/dynamic response.

Each fuel assembly in the NFV is supported vertically on a support casting that essentially rests on the NFV floor. Consequently, the small reductions in the vertical natural frequencies do not significantly change the vertical seismic/dynamic response.

The bundle weight of GE14 fuel with 100/65/50 Mil channel is 646 lbs. Therefore, the NFV has been qualified for a fuel design (i.e., GE6-80 at 666 lbs per bundle) with a 3.00% greater per bundle weight than that of the GE14 fuel. As discussed above, the seismic/dynamic input motion to the NFV does not change with the introduction of GE14 fuel.

Therefore, the introduction of GE14 fuel does not significantly change the dynamic response in the NFV when subjected to the HCGS licensing design basis seismic/dynamic loads. Thus, the HCGS NFV is dynamically qualified for GE14 fuel.

8.2.3. High Density Spent Fuel Storage Racks Seismic/Dynamic Qualification

The High Density Spent Fuel Storage Racks (HDSFSRs) were qualified for all fuel designs for which the plant has been in Commercial Operation. In particular, the original design was based on GE6-80 mil fuel with a per bundle weight of 666 lbs.

The weight of GE14 fuel with a 100/65/50 Mil channel is 646 lbs per bundle. Therefore, the HDSFSRs have been qualified for a fuel design (i.e., GE6-80 with a per bundle weight of 666 lbs.) with a 3.00% greater per bundle weight than that of the GE14 fuel.

As discussed in the incore evaluation above, the results of actual HCGS horizontal primary structure eigen analyses, shows that there is minimal change in the fuel frequency for both cases, i.e., GE14 fuel and the mixed core of GE14 and SVEA-96+ fuel. For the HCGS HDSFSRs, the less than 3.39% reduction in the horizontal fuel assembly natural frequencies, due to the introduction of GE14 fuel, does not significantly change the HDSFSRs seismic / dynamic response.

Each fuel assembly in the HDSFSRs is supported vertically on a support casting or plate that essentially rests on the spent fuel pool floor. Consequently, the small reductions in the natural frequencies in the vertical direction, do not significantly change the vertical seismic / dynamic response.

Therefore, the introduction of GE14 fuel does not significantly change the dynamic response in the HDSFSR when subjected to the HCGS licensing design basis seismic / dynamic loads. Thus, the HCGS HDSFSR is dynamically qualified for GE14 fuel.

8.3 NEUTRON FLUENCE

Currently, the HCGS core consists mainly of fuel type SVEA-96+. The introduction of GE14 to the HCGS core is assessed here. Various parameters relevant to fast neutron flux were evaluated based on the fuel differences between the transition cycles that will be comprised of the SVEA-96+ and GE14 fuel designs and a full core of GE14. The full core GE14 evaluation is based upon operation for 32 EFPY, with 12 EFPY at 3293 MWt, the original licensed thermal power, 3 EFPY at 3339 MWt, the current licensed thermal power, and the remaining 17 EFPY at 3952 MWt which bounds operation at the current licensed thermal power through the end of the 40-year operating license.

Among the factors affecting neutron flux distribution, the most influential to the flux level at the RPV are core loading pattern, moderator density, fuel mass, and power distribution.

- The core loading pattern will not change due to new fuel introduction. The proximity of bundle location relative to the vessel has a significant effect on the flux level at the RPV. Several peripheral bundles dominate the peak flux at the RPV. Ten of the "most important" bundles contribute to nearly 90% of the RPV peak flux in each octant. These bundles are referred to as 'R1', and are shown in Figure 8-1. The introduction of new fuel products will not affect the relative importance of these bundles. In actual plant operation, a core load asymmetry introduced by plant-unique operations may cause the RPV peak flux to shift from one quadrant to another. However, this type of deviation is

independent of new fuel introduction, and the effect is bounded by the uncertainty and bias of the fluence evaluation processes.

- Moderator density or void distribution can significantly alter the neutron flux distribution in the reactor. However, these parameters are controlled by the operating strategy and the change of these parameters is not directly related to the introduction of new fuel product lines.
- The fissile material content in each fuel node may vary from one fuel design to another. Variations of nodal mass distribution may cause shifting of the peak flux elevation. One of the most distinctive features of GE14 fuel that differ from SVEA-96+ is the part length rod design. Because of the presence of part length rods in GE14, fission neutrons are produced at a higher rate in the lower portion of the core. Meanwhile, fast neutron flux in and around the upper core is reduced. Consequently, flux distribution in a GE14 core tends to peak near the midplane rather than peaking toward the upper core, as in a traditional full-length core. The magnitude of peak flux is not expected to increase due to this shift.
- The average relative power densities generated in R1 bundles in a full core of GE14 fuel versus those of the transition cycles were compared. This comparison shows that the transition cycle energy distributions gradually approach that of a full core of GE14. All of the transition cycles will operate at rated powers that are less than the power level used for the full core GE14 evaluation (3952 MWt). Therefore, the peak fluence at the RPV based on the full core GE14 evaluation bounds the fluence of the mixed fuel transition cycles.

Based on the above, it is concluded that the introduction of GE14 fuel at HCGS will not adversely affect the RPV fast neutron fluence level.

8.4 FUEL HANDLING ACCIDENT

The effect to the radiological consequence of an FHA due to the introduction of GE14 at HCGS was assessed. The results follow:

1. The HCGS licensing basis FHA analysis assumes that 124 fuel rods in an 8x8 array assembly are damaged during the FHA. There are 62 fuel rods per assembly.
2. The GESTAR II GE14 Compliance Document (Reference 2) shows that during a GE14 FHA the maximum number of damaged fuel rods is 172, and includes all the rods in the dropped assembly. It was also assumed there are 92 rods per assembly. Because GE14 contains 14 part length rods, the equivalent number of full-length rods is 87.33 rods per assembly (Reference 2). Therefore, no more than 168 rods are damaged in the GE14 FHA (172-92+87.33 = 167.33 or 168 equivalent full length rods). Both the HCGS FHA licensing basis and the Reference 2 analyses include the weight of the dropped assembly plus the total mass of the telescoping mast and grapple head.
3. Section 8.5 (Source Term) of this report concludes that there is no adverse effect on the HCGS radioactivity inventory due to introduction of GE14. Therefore, the severity of radiological consequence of an FHA can be assessed by comparing the following parameters:

    a. The equivalent number of damaged bundles during the accident.

b. The radial power peaking factors.

4. The equivalent number of damaged bundles in the current analysis is (124 / 62) = 2.0 bundles. Whereas the equivalent number of damaged bundles for the same accident for GE14 is (168 / 87.33) = 1.92 bundles.
5. The radial peaking factor assumed for radiological consequence determination in the HCGS FHA analysis of record (radial peaking factor = 1.75) has been confirmed to bound the expected radial peaking associated with mixed cores of SVEA-96+ and GE14 and full cores of GE14 (Reference 2). As a result, the radiological consequence of an FHA at HCGS involving GE14 fuel only is conservatively determined to be (1.92*1.75) / (2.0 * 1.75) = 0.96 of the current licensing basis. Therefore, the current HCGS licensing basis FHA bounds a FHA accident involving GE14.
6. The current HCGS licensing basis states that the SVEA-96+ fuel is bounded by the original HCGS licensing basis.

Based on the above, a FHA involving a mix of SVEA-96+ / GE14 fuel is also bounded by the HCGS licensing basis.

8.5 RADIATION SOURCE TERM

Currently, the HCGS core load consists mainly of SVEA-96+ fuel. The current design basis source term was generated assuming a 3.0% to 3.9% U-235 enrichment with an EOC core average exposure of 33.9 GWd/MT and a mean discharge exposure that does not exceed 47.5 GWd/MT.

For the initial transition cycle of GE14 at HCGS, the fuel parameters are bounded by the current design basis, i.e., for the initial GE14 reload core the fuel enrichment is approximately 3.64%, the EOC core average exposure is less than 30.0 GWd/MT, and the mean discharge exposure is less than 43 GWd/MT.

Therefore, the initial transition cycle of GE14 will not adversely affect the radiological source term at HCGS. Subsequent GE14 transition cycles up to and including a full core of GE14 fuel at CLTP will also remain bounded by the current source term as long as the enrichment and exposure assumptions remain valid.

Therefore, the initial transition cycle of GE14 does not have an adverse affect on the radiological source term at HCGS.

8.6 HYDROGEN GENERATION/RECOMBINATION ANALYSIS

Analysis of the HCGS Containment Atmosphere Control System (CACS) confirms that the GE14 based analysis is bounding for SVEA-96+ or for any mixed core of SVEA-96+ and GE14 fuel for thermal power levels up to 120% of OLTP. The basis for the bounding or conservative result is that GE14 fuel produces less hydrogen from metal-water reaction than does SVEA-96+ fuel, resulting in 1) less dilution of containment oxygen produced by post-Loss-of-Coolant Accident (LOCA) radiolysis of reactor coolant, and 2) an earlier CACS start time. Using the Regulatory Guide 1.7 basis for metal-water reaction, GE14 fuel results in post-LOCA generation

of 14.46 lb-moles of hydrogen, as compared to 15.62 lb-moles for SVEA-96+ fuel. In addition, because post-LOCA radiolytic oxygen production is proportional to thermal power, at power levels less than 120% of OLTP, radiolytic oxygen production is bounded by the 120% analysis. Thus, the 120% analysis using GE14 fuel results in an earlier CACS start time to maintain oxygen below the Regulatory Guide 1.7 flammability limit of 5% by volume.

On this basis, no further post-LOCA containment flammability analysis is needed in support of SVEA-96+ fuel.

8.7 FUEL STORAGE CRITICALITY

A GE14 new fuel in-core peak eigenvalue (k_∞) criteria of 1.3392 has been determined, consistent with the in-core eigenvalue limit established for GE14 in the HCGS spent fuel storage racks. This common criterion was established to provide one value for the storage of GE14 in all racks (new and spent fuel) currently in use at HCGS.

The HCGS new fuel Low Density Fuel Storage (LDFS) rack arrays were analyzed with GE14 fuel and shown to have a k_{eff} less than 0.90 for both normal and abnormal configurations. The normal configurations included displacement of stored fuel assemblies within the ranges permitted by the rack manufacturing tolerances at a temperature of 20°C. The abnormal configurations consider the storage of new fuel immersed in non-borated water with and without flow channels at centered and off-center bundle locations at normal and elevated temperatures. In order to comply with 10 CFR 50.68(b) (in the absence of administrative moderator controls) to assure sub-criticality at optimum moderation conditions, a checkerboard loading pattern restriction applies. The optimum moderation scenarios evaluated considered vault flooding with low-density water uniformly distributed outside of the fuel boxes (channels) along with both low and high-density water inside of the fuel boxes. The HCGS new fuel storage racks satisfy the reactivity requirements for limited-location storage conditions with GE14 fuel inserted. A new fuel eigenvalue criterion (1.3392) was established and provided to maintain a consistent basis with the independently determined HCGS GE14 spent fuel eigenvalue criterion and GNF methodology.

The HCGS spent fuel High Density Fuel Storage (HDFS) rack arrays were analyzed with GE14 fuel and shown to have a k_{eff} less than 0.95 for both normal and abnormal (limiting array) configurations (the low density rack array simulations were limited to nominal material/geometry evaluations, given that low density system's large inherent margin to the spent fuel reactivity criteria). The normal configurations included displacement of stored fuel assemblies within the ranges permitted by the rack manufacturing tolerances at a temperature of 20°C. The abnormal configurations considered the storage of spent fuel without flow channels and with off-center bundle locations. The HCGS spent fuel storage racks satisfy the reactivity requirements for all storage conditions with GE14 fuel inserted. The in-core limiting-lattice eigenvalue (1.3392) was established based upon the reactivity response (in-rack) of the reference design basis GE14 fuel bundle (4.90w%) and applies to all three spent fuel storage rack systems currently in use.

8.8 MECHANICAL COMPATIBILITY

To demonstrate compatibility of GE14 fuel assemblies with SVEA-96+ assemblies at HCGS, a design layout was created for SVEA-96+ showing the relevant mechanical interfaces between the adjacent fuel assemblies. This layout addresses the vertical and lateral locations of the channel spacer, the channel fastener spring interface surface, and the channel fastener guard spring stop. The layout shows that there is sufficient lateral overlap between channel spacers, between channel fastener springs, and between channel fastener guardrails of adjacent GE14 and SVEA-96+ fuel.

The design layout shows beginning-of-life vertical overlap between channel spacers of 1.25 inches, a vertical overlap of 1.73 inches between the mating spring contact surfaces, and an overlap of 0.79 inches between the channel fastener spring stop mating surfaces. These overlaps are large enough to accommodate the vertical changes due to irradiation growth between any combination of adjacent GE14 and SVEA-96+ bundles such that sufficient vertical overlap will be maintained. Therefore, GE14 fuel assemblies are mechanically compatible with adjacent SVEA-96+ fuel assemblies.

8.9 EMERGENCY PROCEDURE GUIDELINES

Plant-specific Emergency Operating Procedures (EOPs) are based upon plant-specific data, fuel-specific data, and generic parameters specified in the BWR Owners' Group Emergency Procedure Guidelines (EPGs). In support of plant reloads, data for the GE14 fuel product was determined. The specific values are provided in Table 8-1. For core loadings that include fuel designs other than GE14, a comparison of the values provided in Table 8-1 will be performed for all fuel designs and the appropriate fuel specific data will be used in accordance with HCGS's implementation of the BWR Owners' Group EPGs.

8.10 SLCS MARGIN CRITERIA FOR SVEA-96+ FUEL

Because of the introduction of GE14 fuel into the HCGS core containing SVEA-96+ fuel, the Standby Liquid Control System (SLCS) shutdown margin (SDM) criteria applicable to this alternate vendor fuel product was required to be determined using GE methodologies.

To perform this assessment, TGBLA06 and MCNP Monte Carlo analyses of representative SVEA-96+ lattices present in the HCGS core were analyzed in both the cold uncontrolled unborated and borated states at beginning of life. The analysis temperature for this condition was 160°C and the borated concentration was 660 parts per million (ppm). From this information, the boron worths in units of both Δk and $\Delta k/k$ were calculated and compared to the boron worths for the GNF 10x10 fuel data used in the development of the GNF 10x10 SLCS SDM criteria for TGBLA06/PANAC11. The results of this comparison showed that the SVEA-96+ lattices have a slightly negative difference between the TGBLA06 and MCNP boron worths ($\Delta k_{BORON\text{-}TGBLA06} - \Delta k_{BORON\text{-}MCNP}$), which indicates that TGBLA06 slightly under predicts the worth of the boron in the SVEA-96+ fuel (by about -0.002 to -0.003 Δk for boron worths of about 0.110 to 0.120 Δk). For GNF 10x10 designs, TGBLA06 over predicted the boron worth on the average by a very small amount (by about 0.0 to 0.001 Δk for boron worths of about 0.13 Δk). A two-sample T-test comparing the SVEA-96+ lattice results to the GNF lattice

results showed that they were not of the same population. While not of the same population (because of the separation of the means), the standard deviation of the data for both fuel types is quite similar.

Because the standard deviations are so similar and the SVEA-96+ lattice TGBLA06 results under predict the boron worths slightly relative to MCNP, it is acceptable and slightly conservative to apply the GNF 10x10 SLCS SDM criteria to the SVEA-96+ fuel product. Because of the limited lattice data used in this study, this conclusion applies only to the specific SVEA-96+ bundle designs being used in the HCGS cores at the time of the introduction of GE14.

8.11 BPWS ACCEPTABILITY

The rod drop evaluations performed [[ ]] on a representative high-energy equilibrium GE12 and GE14 core designs for Banked Position Withdrawal Sequence (BPWS) plants as described in the current revision of NEDC-32868P (Reference 2) were determined to be applicable to HCGS.

The applicability bases are described in Reference 2. The evaluations for GE14 compliance to the BPWS analyses show that the CRDA is eliminated as a safety concern by:

1. Verifying that criticality does not occur prior to the withdrawal of the first Control Rod of the second BPWS group.
2. Determining the rod worth for control rods in the center region of the core at hot startup conditions and comparing them with the [[ ]] values.
3. Determining the rod worth for control rods at the periphery of the core and comparing them with the [[ ]] values.
4. Comparing the Doppler Coefficients with the [[ ]] values.
5. Comparing the local peaking factors with the [[ ]] values
6. All control rod worths were less than the confirmation criterion of 1% Δk.

Because HCGS is a BPWS plant and follows a [[ ]] Control Rod Drop Accident (CRDA) withdrawal sequence, [[ ]]. Even though the core will contain SVEA-96+ and GE14 fuel designs for several cycles, the enrichment of SVEA-96+ is lower than the enrichment for GE14. Therefore, as long as HCGS follows the BPWS rules, the presence of SVEA-96+ fuel would not invalidate the [[ ]] analysis to HCGS.

In conclusion, the GE14 design for HCGS passes the CRDA compliance check for the continued applicability of the BPWS analysis.

Table 8-1
GE14 Data for Emergency Operating Procedure Guidelines

Parameter	*Value*	
Cold shutdown boron concentration requirement for naturally occurring boron	660 ppm	
Hot shutdown boron concentration requirement for naturally occurring boron	755 ppm [1] 522 ppm [2]	
Minimum active fuel length fraction which must be covered to maintain PCT < 1500°F with injection	83.33% for LPF [3] ≤ 1.4 [4]	
Minimum active fuel length fraction which must be covered to maintain PCT < 1800°F without injection	70.83% for LPF [c] ≤ 1.4 [d]	
Minimum bundle steam flow required to maintain PCT < 1500°F	1340 [5] lbm/hr	
Maximum core uncovery time before PCT exceeds 1500°F for peak linear heat generation rate of 13.4 kW/ft (t-cu-15)	Time after Shutdown (minutes)	t-cu-15 (minutes)
	1	1.28
	5	1.73
	10	2.01
	30	2.67
	50	3.07
	80	3.61
	100	3.90
	500	5.58
	1000	6.76
	3000	9.26
	6000	11.76
Specific heat of clad at 500°F	0.0761 Btu/lbm-°F	
Specific heat of fuel at 500°F	0.0683 Btu/lbm-°F	
Fuel Components Mass [6] (lbs)	Channel 100/65	Channel 120/75
	205	216
Fuel Mass assuming no gad (lbs)	456	
Active Fuel Length (inches)	150	

[1] Value specified per the basis identified in EPG Rev. 4. GE14 is covered by the values specified for 8x8 designs.
[2] Value specified per the basis identified in EPG/SAG Revision 1: control rods withdrawn to maximum rod block limit, core pressure 1100 psia saturated liquid, full power equilibrium xenon, no voids in core, no shutdown cooling, and initial reactor condition at Maximum Extended Operating Domain and most reactive exposure. GE14 is covered by the value specified for 8x8 designs.
[3] Local peaking factor (LPF) is defined as the maximum rod-to-rod local peaking factor in the uncontrolled portion of the bundle. The LPF in the controlled portion of a bundle is not relevant because it is not used in the steam cooling calculations.
[4] The LPF value influences the capability of the bundle to produce sufficient steam to keep the uncovered portion of the highest-powered rod cooled to the specified temperature. An LPF value of ≤ 1.4 is acceptable for all possible bundle axial power shapes. This is based on a maximum axial peaking factor (APF) of 2.0. Therefore any combination of LPF and APF such that LPF*APF ≤ 1.4*2.0 is similarly acceptable. An LPF value of ≤ 1.59 is also acceptable for all bundle axial power shapes peaked at or below the midplane for a maximum APF of 2.0.
[5] This value is for a peak linear heat generation rate of 13.4 kW/ft.
[6] This includes all components except for the fuel pellets. The stainless steel components (Upper and Lower tie plates, etc.) are included by calculating an equivalent zircaloy mass based on the ratio of the stainless steel to the zircaloy specific heats.

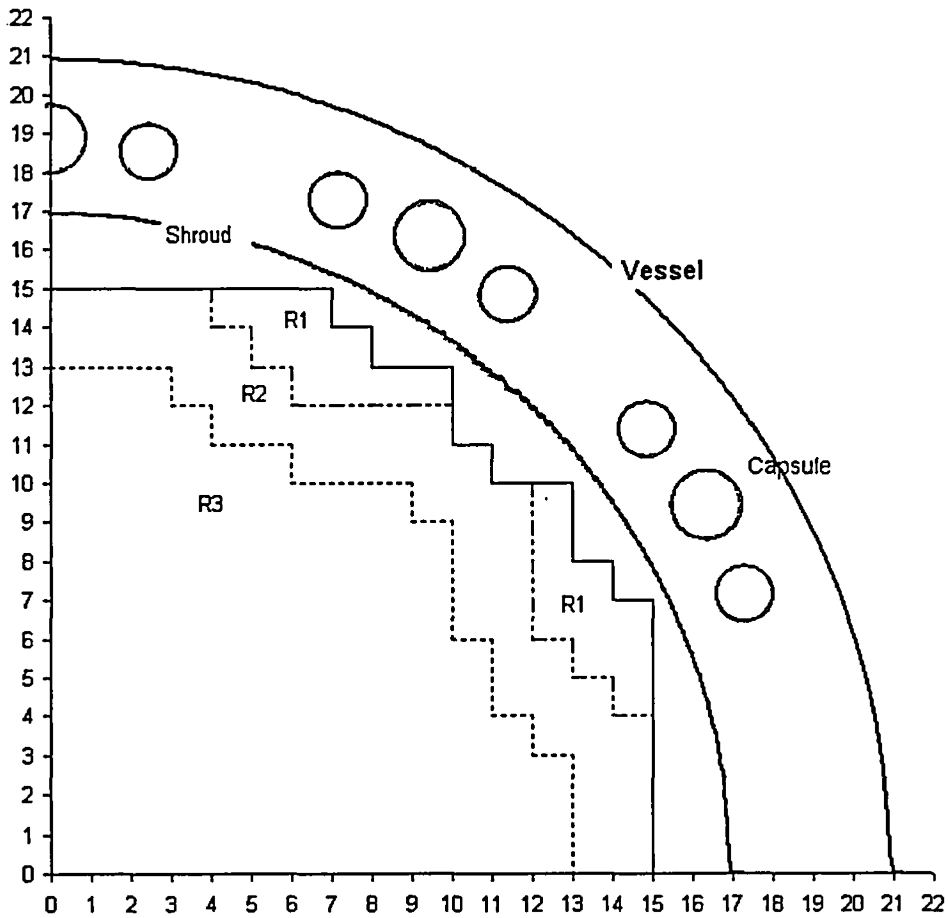


Figure 8-1
The "R1" Bundles

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