NEDO-33158, Supplement 1, Revision 1
Fuel Transition Report for Hope Creek Generating Station

GE Energy, Nuclear

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Fuel Transition Report For Hope Creek Generating Station

Supplement 1

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REVISION

NEDO-33158, Supplement 1, Revision 0 was submitted to PSEG Nuclear LLC, in March 2005. The only change in Revision 1 is the addition of proprietary markings in Section 4. NEDO-33158, Supplement 1, Revision 1 replaces NEDO-33158, Supplement 1, Revision 0 in its entirety.

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ACRONYMS AND ABBREVIATIONS

Term	Definition
AOO	Anticipated Operational Occurrence
ARI	Alternate Rod Insertion
ARTS	Average Power Range Monitor / Rod Block Monitor / Technical Specifications
ATWS	Anticipated Transient Without Scram
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CACS	Containment Atmosphere Control System
CLTP	Current Licensed Thermal Power
°F	Degrees Fahrenheit
ELLLA	Extended Load Line Limit Analysis
EOC	End-of-Cycle
EPG	Emergency Procedure Guideline
FHA	Fuel Handling Accident
GE	General Electric Company
GNF	Global Nuclear Fuel - LLC
GWd/MT	Gigawatt-Days per Metric Ton
HCGS	Hope Creek Generating Station
LHGR	Linear Heat Generation Rate
LHGRFAC	Linear Heat Generation Rate Adjustment Factor
LHGRFAC (F)	Linear Heat Generation Rate Flow Dependent Adjustment Factor
LHGRFAC (P)	Linear Heat Generation Rate Power Dependent Adjustment Factor
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
NRC	Nuclear Regulatory Commission
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
PCT	Peak Cladding Temperature

Term	Definition
ppm	Parts per million
PSEG	PSEG Nuclear LLC
RBM	Rod-Block Monitor
RCIC	Reactor Core Isolation Cooling
RIPD	Reactor Internal Pressure Differences
RPT	Recirculation Pump Trip
RSLB	Recirculation Suction Line Break
RTP	Rated Thermal Power
SDM	Shutdown Margin
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-Loop Operation
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake

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 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 reviewed to support the continued use of the GE14 fuel design at Hope Creek Generating Station (HCGS) for a Constant Pressure Power Uprate (CPPU) of 115% of the Current Licensed Thermal Power (CLTP of 3339 MWt) to a CPPU Rated Thermal Power (RTP) level of 3840 MWt. The [[ ]] analyses are performed for each reload regardless of fuel design or licensed power level.

HCGS will continue to utilize GE14 fuel for CPPU operation. Currently, the plant is operating with some 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 supplement to NEDC-33158P summarizes the results of the [[ ]] review of the HCGS mixed core of GE14 and SVEA-96+ at CPPU RTP. The mixed fuel core will consist of approximately 50% GE14 / 50% SVEA-96+. The reviews were performed considering the Maximum Extended Load Line Limit Analysis (MELLLA) operating domain.

All of the topics analyzed in NEDC-33158P are reviewed in this supplement. A confirmatory statement is provided for each topic and describes any changes from the results reported in NEDC-33158P. Many of the topics evaluated in NEDC-33158P were evaluated at CPPU RTP for the introduction of GE14 fuel in Cycle 13 and those evaluations are bounding for CPPU RTP operation. Other topics are independent of power level and therefore require no action for CPPU and it is stated as such herein. For those topics requiring updated results, the updated results are presented with a statement as to their acceptability.

The results of the [[ ]] analyses and evaluations contained in this report conclude that HCGS can safely operate using a mixed core of GE14 and SVEA-96+ fuel at CPPU RTP.

2.0 STABILITY EVALUATION

The generic bases and licensing requirements presented in NEDC-33158P continue to remain applicable to CPPU operation. The stability evaluation for CPPU will be provided separately and will include Backup Stability Protection, DIVOM and OPRM Option III armed region evaluations

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. This assessment has been updated to address operation at CPPU RTP.

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
- CPPU RTP
- Extended Load Line Limit Analysis (ELLLA) and MELLLA power flow operating domains

3.2 CORE DECAY HEAT ANALYSIS

The CPPU core decay heat analyses continue to be based on either the ANS 5.1-1979 Standard or the May-Witt decay heat table. This assessment provides justification that the decay heat bases are not adversely affected when applied to SVEA-96+ and GE14 fuel designs in the CPPU condition.

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. The inputs for this calculation remain applicable for the GE14/SVEA-96+ fuel designs that are utilized for CPPU operation and the comparison results are independent of the power level selected for the comparison. Therefore, the decay heat bases continue to remain valid for CPPU operation.

Because this comparison was made for a conservatively low enrichment and a conservatively long irradiation period and was independent of power level, the decay heat bases remain valid for operation at CPPU RTP.

Justification for the continued use of the May-Witt decay heat 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 May-Witt design and licensing basis will continue to be applicable to CPPU cycles. A comparison of May-Witt with a conservative application of ANS 5.1-1979 for HCGS is shown in Figure 3-1[1]. Although this comparison was developed for the introduction of GE14 fuel at HCGS, it remains valid without change for CPPU application.

For these reasons, the ANS 5.1-1979 Standard and the May-Witt decay heat table continue to provide conservative core decay heat bases for operation at CPPU RTP.

3.3 SPENT FUEL POOL COOLING DECAY HEAT ANALYSIS

The Spent Fuel Pool Cooling (SFPC) analysis for HCGS consists of an assessment of the ability of the Fuel Pool Cooling and Cleanup System (FPCCS) and Residual Heat Removal (RHR) Supplemental Fuel Pool Cooling to meet the design objectives for projected equilibrium cycle batch and core offload scenarios. Two modes of operation are considered when RHR Supplemental Cooling is required: Normal RHR Fuel Pool Cooling Assist mode (RFA) and Alternate RHR Fuel Pool Cooling Assist mode (ARFA).

The following parameters were evaluated for each of the two offload scenarios:

- Spent Fuel Pool (SFP) bulk water temperature.
- Time to boil when no fuel pool cooling system is available.
- Makeup water flow rate.
- Earliest time the ARFA can assume Shutdown Cooling (SDC) and SFPC functions without an RHR train cooling the RPV.

The decay heat basis for CPPU is the ANS 5.1-1979 Standard. Because it has been established that there is no practical difference in decay heat between GE14 and SVEA 96+ for the same enrichment and exposure, the decay heat tables used in the CPPU analysis for SPFC apply equally to mixed cores of GE14 and SVEA 96+.

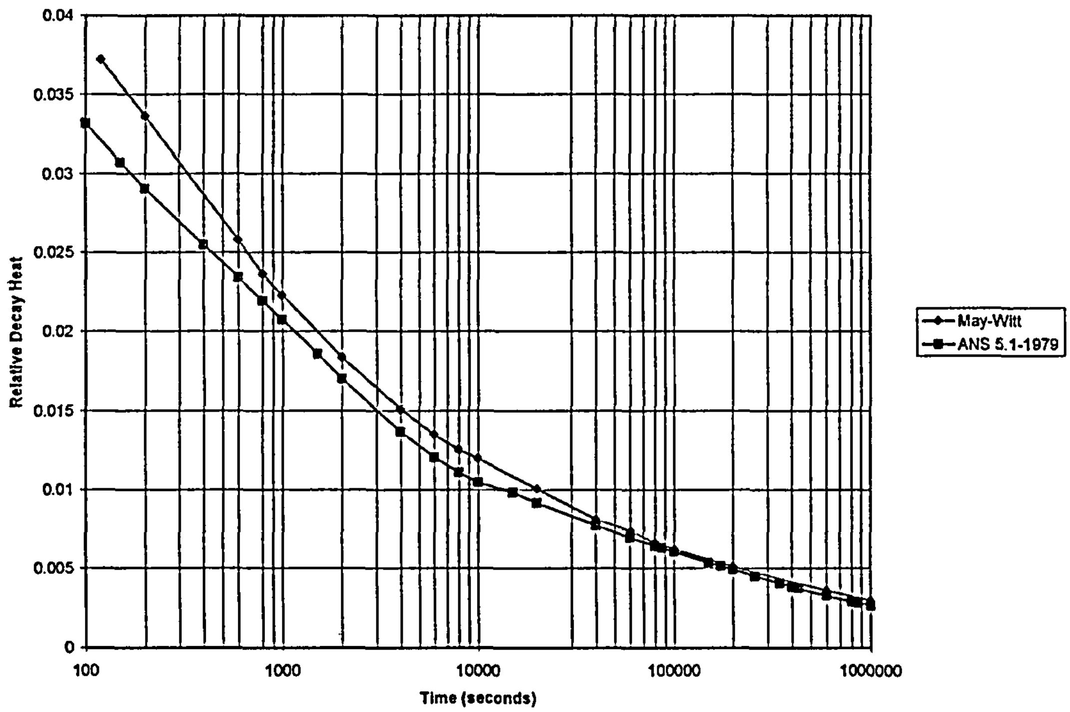
The SFPC analyses are based on a conservative set of assumptions about the loading of the pool. For example, the discharge batch from an equilibrium cycle is used to completely fill the pool, while leaving only enough room for a full core discharge. Although the discharge batch size may be slightly different for SVEA 96+ as opposed to GE14 (which was assumed in the CPPU analysis), the difference in the batch size that might occur is not large enough to affect the conservatism in the SFPC analysis.

3.4 CONCLUSIONS

The CPPU decay heat analysis, although based on a full core of GE14, bounds all transition cycles. Therefore, the power uprate from CLTP to the CPPU RTP, including the consideration of mixed cores of SVEA96+ and GE14, is supported by the CPPU decay heat analyses.

[1] Because the table in the UFSAR includes fuel relaxation energy and the CPPU table does not, the comparison starts at 120 seconds, which is beyond the maximum time for fuel relaxation energy.

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

The Reactor Internal Pressure Difference 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 analysis assumptions in NEDC-33158P, such as applying ICF domain to bound ELLLA/MELLLA domain and considering the lower limit of rated feedwater temperature for pressure differences, fuel lift margin and CRGT lift force calculations, remain unchanged for CPPU RTP operation. The analysis approach and inputs in NEDC-33158P remain the same and are applicable to CPPU RTP operation.

The analysis results in NEDC-33158P remain unchanged and the results based on 120% OLTP bound CPPU RTP operations. For the Normal and Upset conditions at CLTP, the pressure differences across a full core of a single fuel type GE14 or SVEA-96+ bound the transition cores to GE14 except for the pressure differences across the channel walls of SVEA-96+ for which only 1% higher-pressure differences were observed. This 1% difference does not have a significant effect on the channel wall structural integrity as well as the core hydraulic characteristics that can affect the fuel lift margin and CRGT lift force evaluations and is within the calculation uncertainty and conservative basis for determining RIPD, fuel lift margin and CRGT lift force. This same conclusion has been determined to be applicable for mixed cores of SVEA-96+ and GE14 when evaluated at CPPU conditions.

For the Emergency and Faulted RIPD analyses at CLTP, the LAMB code was used to produce the results. The LAMB code is only capable of analyzing a full core of a single fuel type. Therefore, the final Emergency and Faulted RIPD values presented in NEDC-33158P were 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 (i.e., full core vs. mixed core). This same conclusion has been determined to be applicable for mixed cores of SVEA-96+ and GE14 when evaluated at CPPU conditions.

In addition, acoustic and flow-induced loads on jet pump, core shroud and shroud support due to a recirculation suction line break are independent on the fuel type. The analysis results in Reference 4 remain unchanged and are applicable for CPPU RTP operation.

Therefore, for CPPU RTP of 3840 MWt, the current evaluation based on 3952 MWt (120% OLTP) is bounding for the CPPU RTP condition.

5.0 STRUCTURAL ASSESSMENT

The loads associated with the RIPD change with the introduction of a new fuel design. The effect of this 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. Therefore, because the RIPD analysis was performed for 120% OLTP (3952 MWt), the structural assessment is bounding for the CPPU RTP condition of 3840 MWt.

6.0 TRANSIENT ANALYSES

6.1 REACTOR RECIRCULATION PUMP SEIZURE EVENT

The reactor recirculation pump seizure event was analyzed for Single Loop Operation (SLO) at HCGS and 100% CLTP for the introduction of GE14 fuel. This analysis was performed in Reference 4, Section 6.2, for the HCGS Cycle 13 transition cycle with GE14 and SVEA-96+ fuel in the core and transient analysis inputs consistent with the Reload 12/Cycle 13 analyses. The SLO OLMCPR required as a result of this analysis was provided in Reference 4. The Reference 4 analysis was performed generically with sufficient conservatism for future cycle applications. In addition, the absolute power and flow basis does not change with CPPU. However, if the cycle specific SLO SLMCPR changes then the SLO OLMCPR may be adjusted. The SLO pump seizure based OLMCPR adjustment for HCGS at 115% CLTP is provided in the cycle specific SRLR.

6.2 OFF-RATED POWER AND FLOW DEPENDENT LIMITS

The potentially limiting anticipated operational occurrences (AOOs) and accident analyses were evaluated to support HCGS operation with ARTS off-rated limits as well as operation at CPPU RTP. Analyses were performed to determine the limiting MCPR requirement based on the HCGS fuel and core configuration at CPPU and the off-rated power and flow dependent MCPR and LHGRFAC limit curves.

The off-rated power dependent MCPR limits curve is provided in Figure 6-1 and the off-rated power dependent LHGRFAC limits curve is provided in Figure 6-2. These curves do not contain below Pbypass actual absolute OLMCPR values because the percent power threshold for thermal margin monitoring and Pbypass are the same. The extension of the K(P) limit to the Pbypass power level was confirmed with the limiting transient analysis at the Pbypass power level.

The off-rated flow dependent MCPR limits curve is provided in Figure 6-3 and is consistent with the generic ARTS curve based on a maximum recirculation system runout flow of 109.0%. The off-rated flow dependent LHGRFAC limits curve is provided in Figure 6-4 and is consistent with the curve provided in the Reference 1.

6.3 EFFECTS OF THE POWER LOAD UNBALANCE AT OFF-RATED POWER

The issue identified in Reference 2 has been evaluated for HCGS. Reference 2 identified an inconsistency between the performance of the turbine protection systems and the transient analysis assumptions for a generator load rejection event. In particular, in the operating domain between Pbypass and the point at which the Power Load Unbalance (PLU) system is enabled, the response to a generator load rejection would be a slow closure of the turbine control valves (TCVs). The transient analysis assumes a TCV fast closure, which would initiate a reactor scram. Above the PLU enabling power level, the TCV fast closing function will occur. For HCGS, a generator load rejection below the PLU power level would generate a delayed (by ≤ approximately 1 second) turbine trip, which would cause a scram on turbine stop valve position. An analysis was performed with the delayed scram at 55% of CPPU to bound the ~47.3% PLU power level. The analysis results showed that the generic K(P) and LHGRFAC(P) limits bound this event in the range between Pbypass and the PLU enabling power level.

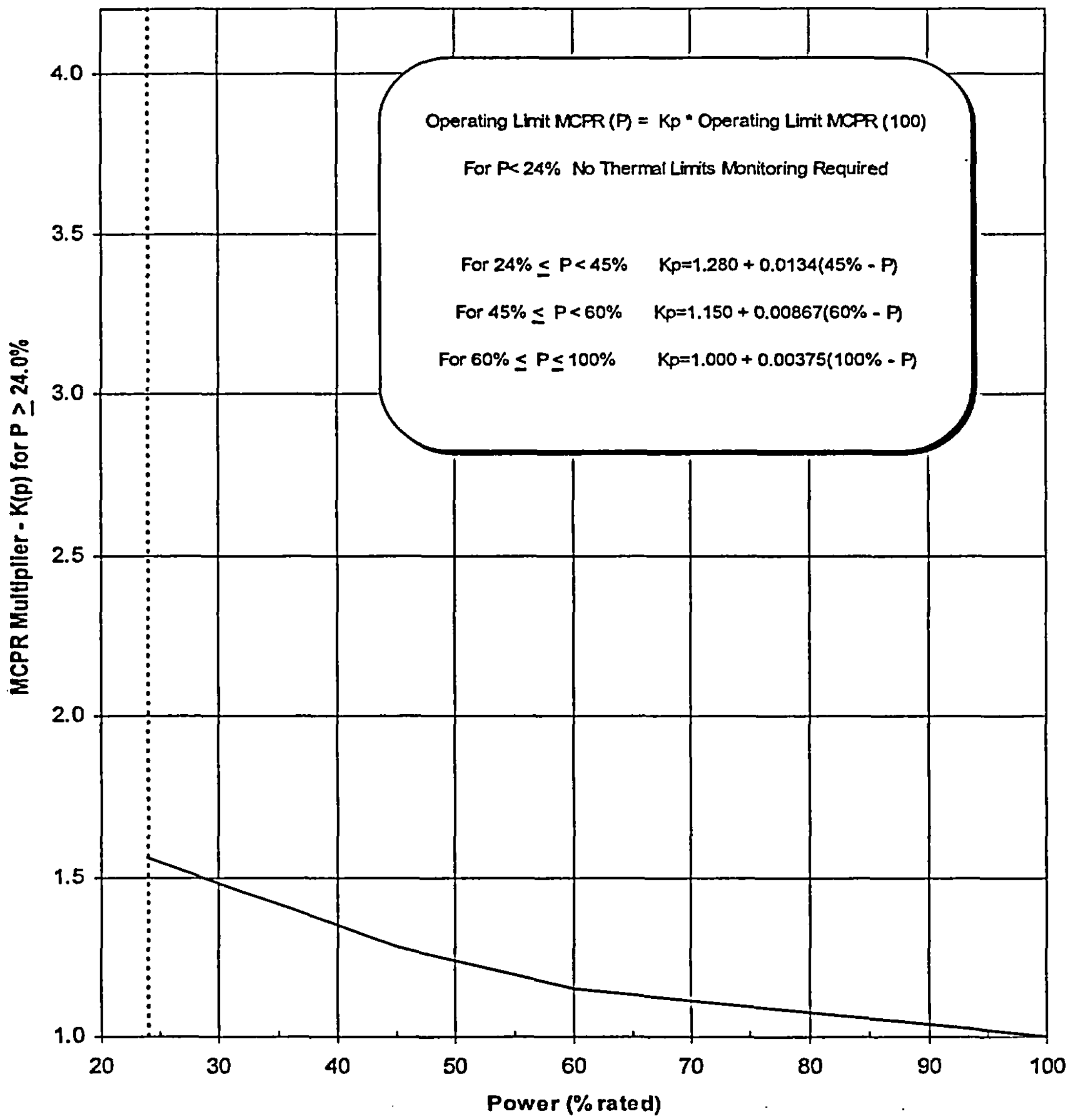


Figure 6-1
Power-Dependent Limit, K(P)

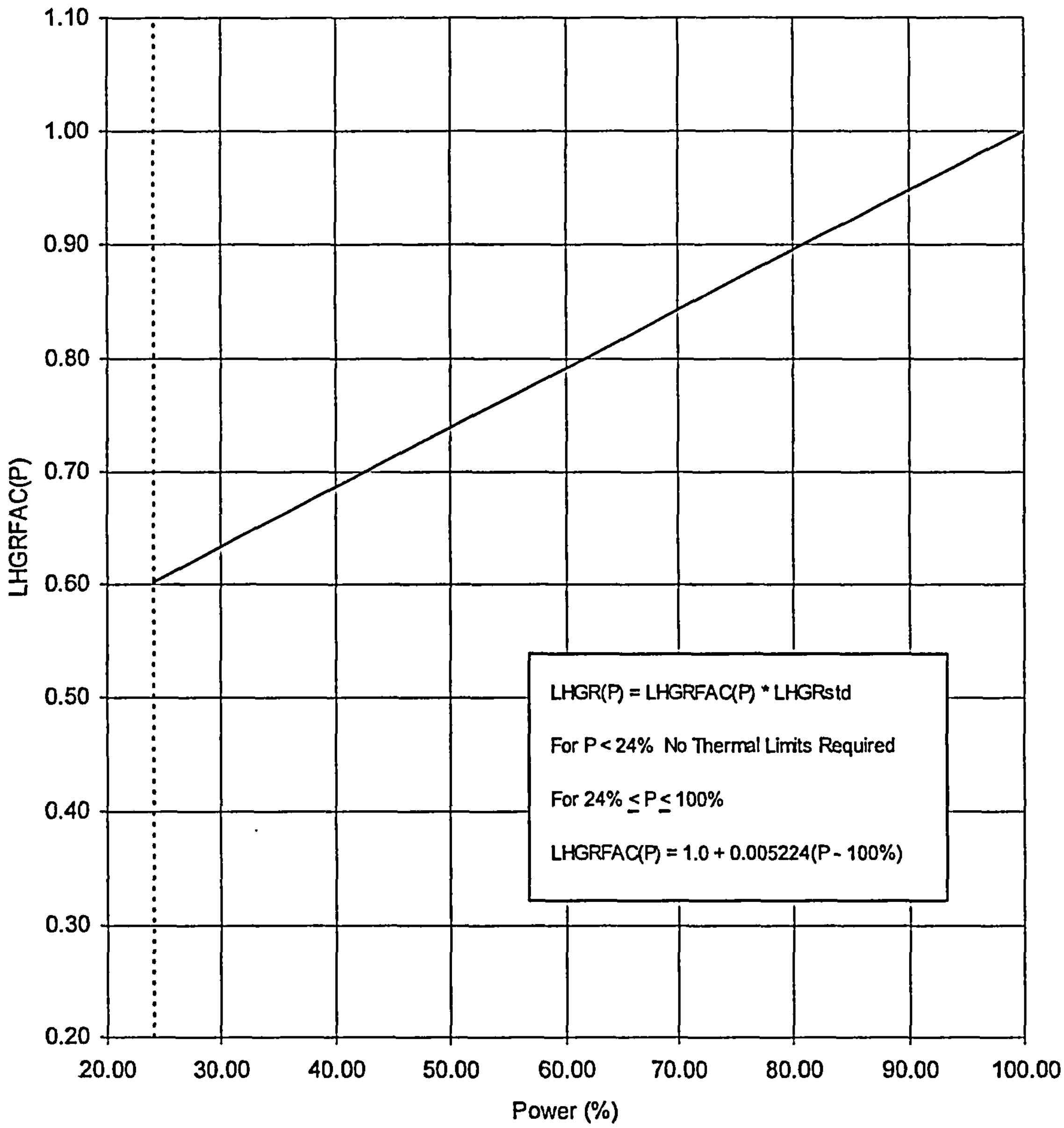


Figure 6-2
Power-Dependent Limit, LHGRFAC(P)

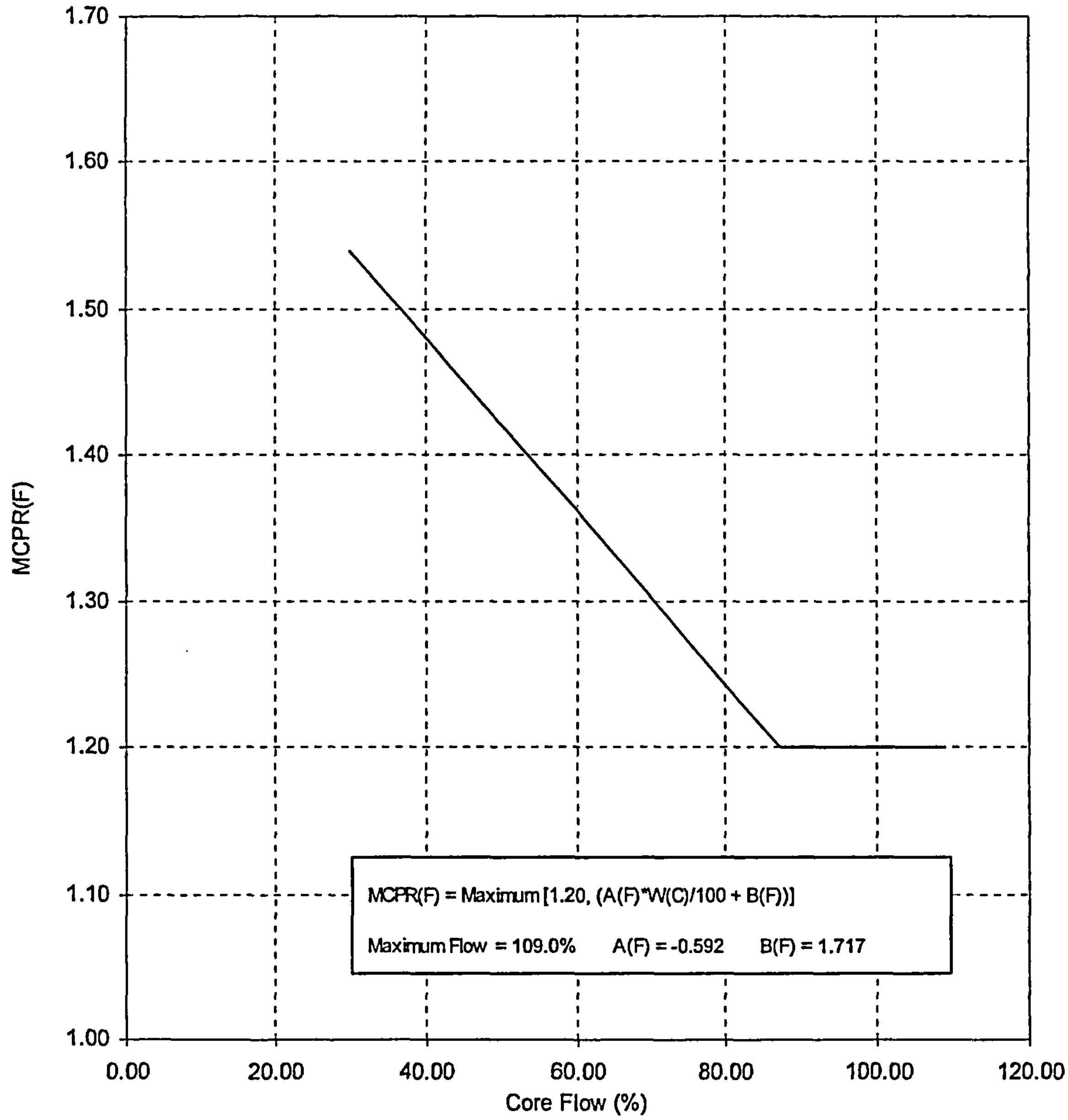


Figure 6-3
Flow-Dependent MCPR Limits, MCPR(F)

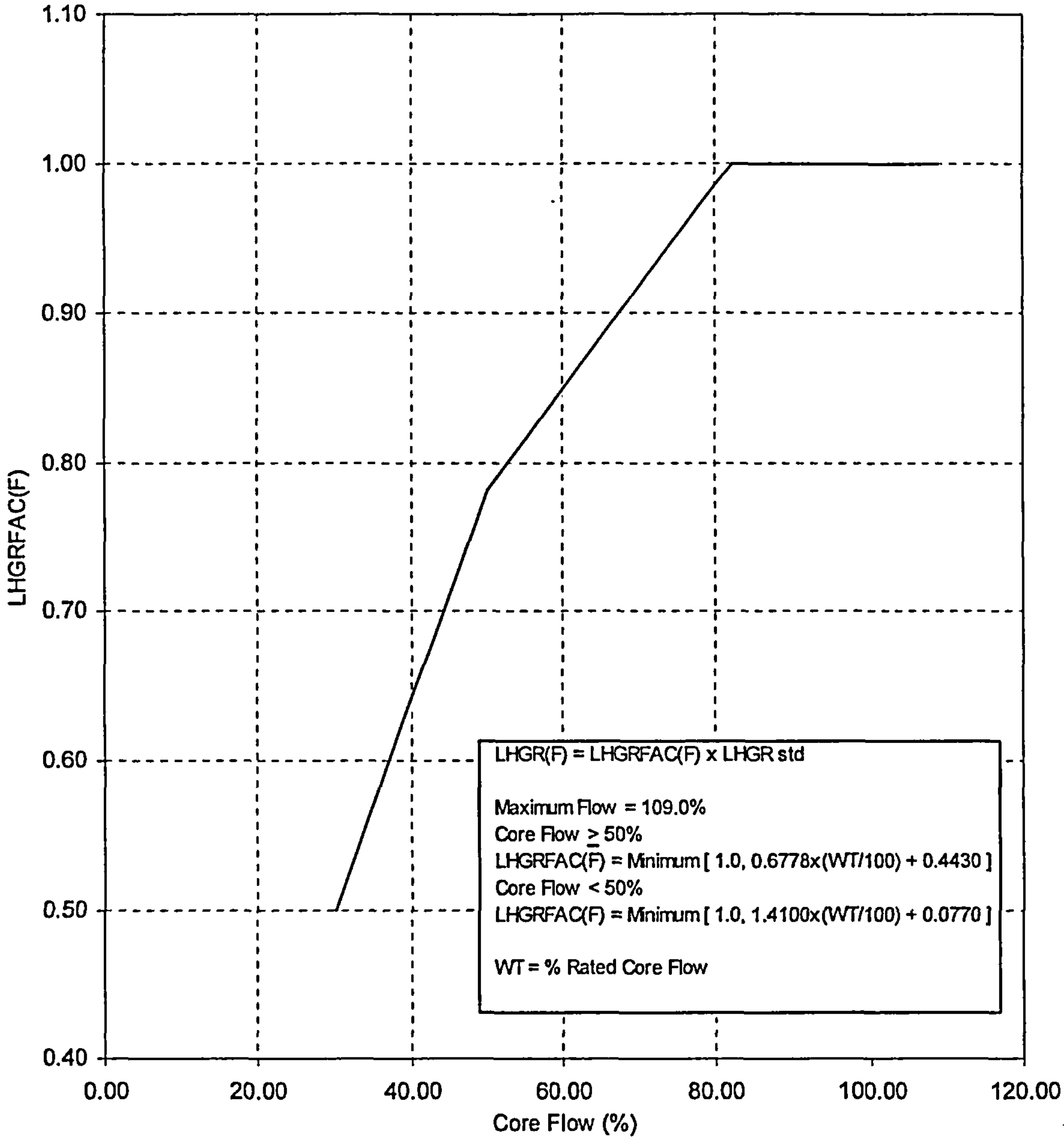


Figure 6-4
Flow-Dependent LHGR Multiplier, LHGRFAC(F)

7.0 APPENDIX R

NEDC-33158P reported that GE14 is the bounding fuel in terms of the initial void fraction. The initial void fraction is the governing parameter affecting the water level and peak cladding temperature (PCT) response during a HCGS Appendix R event where the Reactor Core Isolation Cooling (RCIC) system is utilized to maintain the water level above the top of active fuel. An initial void fraction of approximately 48% was determined for a full core of GE14 at CLTP, which bounds the initial void fractions for both the transition core of GE14 and SVEA-96+ and a full core of a single fuel type of SVEA-96+. This bounding relationship has been determined to be applicable for mixed cores of SVEA-96+ and GE14 at CPPU conditions. Consequently, the GE14 fuel was also assumed for the CPPU analysis. Therefore, the GE14 results in the CPPU analysis are applicable for the CPPU RTP mixed core operation.

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 HCGS 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	(2200°F)
Cladding Oxidation	(<17%)
Peak Suppression Pool Temperature	(201°F)
Peak Containment Pressure	(62 psig)

The results of a plant-specific ATWS evaluation for HCGS CPPU based on a mixed core of GE14 and SVEA fuels indicate that deployment of GE14 in the core does not adversely affect the plant capability to meet the ATWS acceptance criteria. This evaluation includes the effects of 1 safety relief valve (SRV) out-of-service (OOS) and operation at CPPU RTP in the MELLLA domain.

Results for Limiting Events for CPPU ATWS Analysis

Event	Exposure	Peak Vessel Pressure (psig)	Peak Cladding Temperature[1] (°F)	Peak Suppression Pool Temp[2] (°F)	Peak Containment Pressure (psig)
MSIVC	BOC	1371	1187	-	-
MSIVC	EOC	1370	1397	178	8.4
PRFO	BOC	1387	1363	-	-
PRFO	EOC	1385	1504	177	8.3

Table Notes:

1. Fuel clad oxidation is insignificant (<1%). PCT values reported are the limiting of GE14 and SVEA fuel.
2. The peak suppression pool temperatures are bounded by the corresponding peak value from the CPPU evaluation (Reference 3). The peak suppression pool temperatures after depressurization are also bounded by the CPPU results and meet the acceptance criteria.

8.2 SEISMIC EVALUATION

The seismic evaluation documented in NEDC-33158P was performed to evaluate the effect of the introduction of GE14 fuel. The evaluation addressed mixed cores of SVEA-96+ and GE14 fuel up to and including a full core of GE14 fuel. The seismic evaluation is not power dependent and therefore the evaluation provided for the introduction of GE14 fuel remains valid for subsequent core loads independent of CPPU.

8.3 NEUTRON FLUENCE

Neutron fluence is a key input for reactor vessel fracture toughness evaluations. HCGS has been evaluated for CPPU RTP conditions relative to reactor vessel fracture toughness considerations in Reference 3, Section 3.2.1. The evaluations and conclusions presented in Reference 3 are based upon a full core of GE14 fuel but are bounding and valid for CPPU conditions independent of the fuel design used.

For CPPU RTP operation, HCGS will utilize mixed cores of SVEA-96+ and GE14 through full cores of GE14. Various parameters relevant to fast neutron flux were evaluated to address CPPU RTP and SVEA-96+ / GE14 mixed core conditions. As part of the supporting analysis for neutron fluence in Reference 4, CPPU transition cycles of GE14 and SVEA-96+ fuel through a full core of GE14 have been evaluated based on the projected cycle-specific core designs from GNF's fuel cycle analysis for the GE14 new fuel introduction. The result of this evaluation indicates that the conclusions presented in Reference 3 based upon a full core of GE14 fuel are bounding and valid for CPPU conditions including consideration of mixed cores of GE14 and SVEA-96+ fuel.

The most influential factors that can affect the neutron flux at the RPV were evaluated and dispositioned as not adversely affected for the introduction of GE14 fuel at CLTP conditions in Reference 4. The bases for the dispositions documented in Reference 4 are not changed for the core designs required for CPPU RTP operation; whether, the core design is a mixed core of GE14 and SVEA-96+ fuel or a full core of GE14. CPPU RTP loading patterns will not affect the relative importance of the fuel bundles that contribute most to the RPV flux, CPPU RTP core designs will not require new or changed operating strategies that would significantly alter the moderator density or void distribution, the relative peaking performance of the two fuel designs is not accentuated or exacerbated by the core design required for CPPU RTP operation, and the average relative power densities for the bundles that contribute most are not adversely affected by the core designs required for CPPU RTP operation.

Based on the above, it is concluded that the implementation of a CPPU at HCGS will not adversely affect the RPV fast neutron fluence level or the conclusions presented in Reference 3.

8.4 FUEL HANDLING ACCIDENT

The radiological consequences of the fuel handling accident (FHA) are based on three parameters; the number of damaged fuel bundles, the radial power peaking factors, and the radioactivity inventory or source term. The mechanical aspects of the FHA; the number of damaged fuel bundles, are independent of reactor power. The radial power peaking factor analysis of record bounds the radial peaking factors for the CPPU RTP core design utilized for this evaluation. Section 8.5 (Source Term) of this report concludes that the core inventory radiation source term is valid for mixed cores of SVEA-96+ and GE14 as well as full cores of GE14 at CPPU conditions. Therefore, the severity of radiological consequence of an FHA at CPPU RTP is within their applicable regulatory limits whether a full core of GE14 or a mixed core of GE14 and SVEA-96+ is assumed.

8.5 RADIATION SOURCE TERM

The core inventory for operation at CPPU RTP for HCGS as described in Section 8.3 of Reference 3 was based upon a source term generated assuming GE14 fuel, an initial bundle enrichment of ≤ 4.6 wt%, an EOC core average exposure of ≤ 35 GWd/MT, a discharge bundle exposure of ≤ 58 GWd/MT, an initial bundle uranium mass of ≤ 182 Kg, and a bundle average power of 5.75 MWt. Core inventory is a function of fuel enrichment, power density, and bundle exposure. Fuel mechanical design plays minimal role in core inventory evaluation. Therefore, the evaluation based on GE14 is applicable to SVEA-96+.

This source term bases is applicable to HCGS operating at either CLTP or CPPU RTP as long as the values of the parameters delineated in the paragraph above are not exceeded.

8.6 HYDROGEN GENERATION/RECOMBINATION ANALYSIS

The analysis of the HCGS Containment Atmosphere Control System (CACS) performed in NEDC-33158P confirmed 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 (3952 MWt). Therefore, this analysis is valid for the 115% of CLTP CPPU (3840 MWt).

8.7 FUEL STORAGE CRITICALITY

Fuel storage criticality issues are independent of power level because by definition all modes of fuel storage, new, spent and in-core are at shutdown conditions. In addition, the core designs required to support CPPU RTP operation do not require any new fuel design characteristic that would require an update to the criticality evaluations already performed. Therefore, fuel storage criticality is independent of CPPU and the evaluation performed for the introduction of GE14 fuel in Cycle 13 remains valid for subsequent cycles at CPPU RTP.

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 change as a function of the exposures of the adjacent bundles. A bounding case is an end of life (EOL) SVEA96+ bundle adjacent to a fresh GE14 bundle. The maximum bundle growth for SVEA96+ at the EOL exposure of 55 GWd/MT is 0.59 inches. Thus the above 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

The EPG parameters provided during the introduction of GE14 for cycle 13 are not power level dependent. The standby liquid control parameters are determined generically for all plants and do not change with power uprate. The steam cooling parameters are also generically determined and are valid for CPPU conditions. Therefore, the EPG parameters provided are valid for CPPU conditions.

8.10 SLCS MARGIN CRITERIA FOR SVEA-96+ FUEL

The SLCS margin criteria for SVEA-96+ fuel as determined previously in NEDC-33158P are not affected by operation at the CPPU RTP. All inputs, assumption, methodologies and conclusions are the same.

8.11 BPWS ACCEPTABILITY

Banked Position Withdrawal Sequence (BPWS) is a reactor startup and low power condition topic. The acceptability of the BPWS for SVEA-96+ fuel has been determined and the RTP level has no affect on the SVEA-96+ disposition relative to BPWS bases. In addition, no new or modified core/fuel designs are required for CPPU RTP operation that would invalidate compliance with the 6 criteria described in NEDC-33158P Section 8.11 for justifying that CRDA is eliminated as a safety concern.

9.0 REFERENCES

1. NEDC-33066P, "Hope Creek Generating Station APRM/RBM/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)", Revision 2, February 2005.
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