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Proj 710

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U.S Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20852-2738

Attention: Chief, Information Management Branch
Program Management
Policy Development and Analysis Staff

Subject: **Review of GE Nuclear Energy Licensing Topical Report NEDC-33004P, Revision 3, "Constant Pressure Power Uprate" (TAC No. MB2510)**

By Reference 1, the NRC provided GE with a non-proprietary version of the Safety Evaluation (SE) for GE's Constant Pressure Power Uprate Licensing Topical Report (CLTR), NEDC-33004P in order to determine the presence of any proprietary information. GE has completed its review and determined, in several instances, that proprietary information is included in the non-proprietary version of the SE.

Attachment 1 consists of pages from the non-proprietary version of the SE that are annotated to indicate the additional proprietary information that that should be protected from public disclosure. The additional proprietary information to be protected is enclosed in boxes, and redacted and offset by brackets consistent with the exiting format used for non-proprietary version of the SE.

The information indicated in Attachment 1 as proprietary is identified as proprietary in the CLTR, and addressed by the proprietary affidavit therein. That proprietary affidavit remains applicable and documents the basis for the proprietary nature of the information identified in Attachment 1 pursuant to 10CFR2.790.

GE requests that the proprietary information identified in Attachment 1 be withheld from public disclosure consistent with 10CFR2.790 based on the affidavit provided in the CLTR.

If you have any questions, please contact, Jim Harrison at (408) 925-2253 or myself.

Sincerely,

J Harrison for George Stramback

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Project No. 710

Reference:

1. MFN 03-027, Letter from William H. Ruland (NRC) to James F. Klapproth (GE), March 31, 2003, Review of GE Nuclear Energy Licensing Topical Report NEDC-33004P, Revision 3, "*Constant Pressure Power Uprate*" (TAC No. MB2510)

Attachments:

1. Markups of Pages from Non-Proprietary SE Including Revisions

cc: AB Wang (NRC)
JF Harrison (GE/San Jose)
MA Lalor (GE/San Jose)
JF Klapproth (GE/San Jose)

Attachment 1

Markups of Pages from Non-Proprietary SE Including Revisions

2.1 Fuel Design and Operation

For each fuel vendor, use of NRC-approved fuel design acceptance criteria and analysis methodologies assures that the fuel bundles perform in a manner that is consistent with the objectives of Sections 4.2 and 4.3 of the SRP and the applicable general design criteria (GDC) of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A.

Fuel bundles are designed to ensure that:

- the fuel bundles are not damaged during normal steady-state operation and AOOs;
- any damage to the fuel bundles will not be so severe as to prevent control rod insertion when required;
- the number of fuel rod failures during accidents is not underestimated; and
- the coolability of the core is always maintained.

The fuel vendors perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel system design can meet the fuel design limits during steady-state, AOO, and accident conditions.

The effect of the CPPU approach on the fuel and core design and operation is described in the CPPU LTR. [.] Fuel design limits are established [ ] for all new fuel product line designs as a part of the fuel introduction and reload analyses using the approved GESTAR-II process.

The power level above which fuel thermal margin monitoring is required may change with the implementation of the CPPU. The original plant operating licenses set this monitoring threshold at a typical value of 25 percent of rated thermal power. [.] For CPPU, the fuel thermal margin monitoring threshold is scaled down, if necessary, to ensure that the monitoring is initiated [.] A change in the fuel thermal monitoring threshold also requires a corresponding change to the TS reactor core safety limit for reduced pressure or low core flow.

The staff agrees that this section [ ] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.2 Thermal Limit Assessment

GDC 10 of 10 CFR Part 50, Appendix A, requires that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents).

The effect of the CPPU on the minimum critical power ratio (MCPR) safety and operating limits and on the maximum average planar linear heat generation rate (MAPLHGR) and linear heat generation rate (LHGR) limits is discussed in the CPPU LTR. The topics considered include:

- Safety Limit MCPR
- MCPR Operating Limit
- MAPLHGR Limit
- Maximum LHGR Limit

The safety limit minimum critical power ratio (SLMCPR) ensures that 99.9 percent of the fuel rods are protected from boiling transition during steady-state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as the result of an AOO.

The MAPLHGR operating limit is based on the most limiting loss-of-coolant accident (LOCA) and ensures compliance with the ECCS acceptance criteria in 10 CFR 50 46. For every new fuel type, the fuel vendors perform LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload licensees confirm that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

In general, the licensee must ensure that plant operation is in compliance with the cycle-specific thermal limits (SLMCPR, OLMCPR, MAPLHGR, and maximum LHGR) and specify the thermal limits in a cycle-specific COLR as required by Section 5 of the plant TS. In addition, while uprated power operation may result in a small change in average fuel burnup, the licensee cannot exceed the NRC-approved maximum burnup limits. In accordance with Section 5 of the TS, any cycle-specific analyses are performed using NRC reviewed and approved methodologies. Therefore, the staff expects that the licensee will appropriately consider the potential effects of uprated power operation on the fuel design limits, and that the current thermal limits assessment will show that the plant can operate within the fuel design limits during steady-state operation, AOOs, and accident conditions.

The staff agrees that this section [

] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.3 Reactivity Characteristics

The effect of the CPPU approach on the minimum shutdown margin and hot excess reactivity is discussed in the CPPU LTR. The topics addressed in this evaluation are:

- Hot excess reactivity
- Shutdown margin

The higher core energy requirements of a power uprate may affect the hot excess core reactivity and can also affect operating shutdown margins. The general effect of a power uprate on core reactivity, as described in Section 5.7.1 of ELTR-1, is also applicable to a CPPU. Based on experience with previous plant-specific power uprate submittals, the required hot excess reactivity and shutdown margin can typically be achieved for power uprates through the standard approved fuel and core reload design process. Plant shutdown and reactivity margins must meet NRC-approved limits established in GESTAR-II on a cycle-specific basis and are evaluated for each plant reload core, [

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The CPPU reload core design will account for any loss of margin for future cycles. The reload core analysis will ensure that the minimum shutdown margin requirements are met for each core design and that the current design and TS cold shutdown margin will be met. Since the licensee will continue to confirm that the TS cold shutdown requirements will be met for each reload core operation, the staff finds this acceptable.

The staff agrees that this section [

] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.4 Stability

The staff review in the area of reactor stability is conducted to ensure that the requirements of GDC 12 of 10 CFR Part 50, Appendix A, "Suppression of reactor power oscillations," are satisfied.

The CPPU LTR has taken exception to one of the generic guidelines in ELTR2, regarding thermal hydraulic stability. The staff SE on ELTR2, Section 3.2.2, "Long Term Solution," states "The prevention and detection/suppression features of the long term stability solutions are either demonstrated to be unaffected by power uprate or are modified and validated in accordance with the solution methodology." The ELTR2 staff SE requires that the thermal hydraulic stability monitoring and monitoring system be validated in accordance with the generic solution methodology using a representative equilibrium core design and included in the application for EPU. [

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Section 3.2 of ELTR-2 documents interim corrective actions (ICA) and four long-term solution (LTS) stability options: Enhanced Option I-A, Option I-D, Option II, and Option III.

A generic evaluation was performed for the ICAs as documented in Section 3.2.1 of ELTR-2. This generic evaluation is applicable for the CPPU. Interim corrective action stability boundaries are the same in terms of absolute core power and flow. The listed power levels, as a percentage of rated power, are scaled [ ] based on the new uprated power.

For the long-term solution options, evaluations are reload core dependent and are performed for each reload fuel cycle. The analyses of each long-term option are addressed in the CPPU LTR. The topics addressed in this evaluation are:

- Enhanced Option I-A
- Option I-D

- Option II
- Option III (OPRM armed region and trip/Hot channel oscillation magnitude)

2.4.1 Plants with Enhanced Option I-A

The stability regions and associated trip setpoints may change with CPPU. Enhanced Option I-A (E1A) is classified as a prevention solution. Plants with the E1A stability solution have analytically-based flow biased APRM flux trip functions (exclusion and restricted regions) and an administratively controlled monitored region that are expressed as a percent of rated power. These features are either confirmed or adjusted for each plant reload. The trip function settings and monitored region for the CPPU will be established by the [ ] analysis that incorporates the uprated power level.

The staff agrees that this section [ ] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.4.2 Plants with Option I-D

The exclusion region may change and SLMCPR protection may be affected by the CPPU. Option I-D is a solution combining prevention and detect-and-suppress elements. The prevention portion of the solution is an administratively controlled exclusion region. The detect-and-suppress feature is a demonstration that regional mode reactor instability is not probable and that the existing flow-biased flux trip provides adequate SLMCPR protection for events that initiate along the rated rod line. These features will be analyzed for the [ ] analysis that incorporates the new rated power level.

CPPU will also affect the SLMCPR protection confirmation. Changes to the nominal flow-biased APRM trip setpoint or the rated rod line require the hot bundle oscillation magnitude portion of the detect-and-suppress calculation to be recalculated. This calculation is not dependent upon the core and fuel design. However, the SLMCPR protection calculation is dependent upon the core and fuel design and is performed for each reload. These features will be analyzed for the [ ] analysis that incorporates the new rated power level.

The staff agrees that this section [ ] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.4.3 Plants with Option II

The exclusion region may change and SLMCPR protection may be affected by CPPU. Option II is a detect-and-suppress solution, which applies to the two BWR/2 plants designed with a quadrant-based APRM trip system. This quadrant-based system will detect either core-wide or regional mode instability. These features will be analyzed for the [ ] analysis that incorporates the uprated power level.

The staff agrees that this section [

] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.4.4 Plants with Option III

The Option III trip setpoint may be affected by CPPU operating conditions. The OPRM armed region will be rescaled with CPPU. Option III is a detect-and-suppress solution, which combines closely spaced LPRM detectors into "cells" to effectively detect any mode of reactor instability. Evaluation is dependent upon the core and fuel design and is performed for each reload The generic analyses for the Option III hot channel oscillation magnitude and the OPRM hardware were designed to be independent of core power. [

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The staff agrees that this section [

] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.5 Reactivity Control

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The scram, rod insertion and withdrawal functions of the CRD system depend on the operating reactor pressure and the pressure difference between the CRD system hydraulic control unit (HCU) and the reactor vessel bottom head pressure. The CRD system was generically evaluated in Section 5.6.3 and J.2.3.3 of ELTR1 and in Section 4.4 of Supplement 1 to ELTR2. The [ ] evaluation concluded that the CRD systems for BWR/2-6 plants are acceptable for EPU as high as 20 percent above the original rated power. Therefore, no additional plant-specific calculations are required beyond confirmatory evaluation.

The topics considered in this section are:

- Scram Time Response (BWR/6 and BWR/2-5)
- CRD Positioning
- CRD Integrity

2.5.1 Control Rod Scram

In pre-BWR/6 plants, the scram times may be decreased by the transient pressure response. [

.] The pre-BWR/6 plant generic scram times for American Society of Mechanical Engineers (ASME) overpressure protection and critical power ratio pressurization transient analyses may not be adversely affected by the reactor transient pressure. Thus, the analyses and results would remain valid

For BWR/6 plants, the increase in the transient pressure response tends to increase the scram time. Because the normal steady-state reactor dome pressure for the CPPU does not change, the scram time performance relative to pre-power uprate plant operation may [.] The BWR/6 design generic scram times for ASME overpressure protection and AOO analyses are based on generic reactor pressure versus time envelopes. The overpressure evaluation described in Section 3.1 of the CPPU LTR will be used to confirm that the transient reactor pressures remain within the generic envelopes.

In addition, scram time testing verifies the scram time for individual control rods. [.] The staff SE for ELTR2 states that "the plant-specific submittal for BWR/6 plants must provide assurance that the scram insertion speeds used in the transient analyses are slower than the requirements contained in the plant."

The staff agrees that this section [ ] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

2.5.2 Control Rod Drive Positioning

The increase in reactor power at the CPPU operating condition results in [.] The automatic operation of the system flow control valve maintains the required drive water pressure [.] The normal CRD positioning function is an operational consideration and is not a safety-related function.

2.5.3 Control Rod Drive Integrity Assessment

GENE indicated that the postulated abnormal operating condition for the CRD design assumes a failure of the CRD system pressure-regulating valve that applies the maximum pump discharge pressure to the CRD mechanism internal components. This postulated abnormal pressure bounds the ASME reactor overpressure limit. [.] In its response to the staff's RAI dated December 18, 2001, (Reference 4), GENE indicated that in those cases where the existing design basis conditions do not bound CPPU conditions, a plant-specific evaluation of the CRD mechanism will be performed to account for other applicable design basis mechanical loads resulting from the reactor vessel motion.

On the basis of its review, the staff agrees with GENE's approach that confirmation of bounding existing design basis or plant-specific evaluations accounting for design basis mechanical loads affecting CRDMs would provide the basis to ensure that the CRDMs meet design basis and performance requirements at CPPU conditions.

RHR system, and the automatic depressurization system (ADS). The following topics are addressed:

- High Pressure Coolant Injection
- High Pressure Core Spray
- Core Spray or Low Pressure Core Spray
- Low Pressure Coolant Injection System
- Automatic Depressurization
- ECCS Net Positive Suction Head

4.2.1 <u>High Pressure Coolant Injection</u>

The increase in decay heat changes the response of the reactor water level following a small break LOCA or a loss of feedwater transient event. There is no change to the normal reactor operating pressure or to the SRV setpoints. The HPCI system, utilized in all BWR/4 and some BWR/3 plants, is designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the reactor vessel. In this event, the HPCI system maintains reactor water level and helps depressurize the reactor vessel. Although for this analysis, the HPCI system is typically assumed to be out-of-service, the adequacy of the HPCI system is demonstrated by the margins discussed in Section 4.3 of the CPPU LTR.

In addition, the HPCI system serves as a backup to the RCIC system to provide makeup water in the event of a loss of feedwater flow transient, as described in Section 9.1 of the CPPU LTR. The adequacy of the HPCI system to meet the safety requirement following a loss of feedwater flow event is discussed in Section 9.1.3 of the CPPU LTR.

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The staff agrees that this section [

] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

4.2.2 <u>High Pressure Core Spray</u>

The HPCS system (with other ECCS systems as backup) is designed to maintain reactor water level inventory during small and intermediate-break LOCAs, isolation transients and LOFW. The HPCS system is designed to pump water into the reactor vessel over a wide range of reactor operating pressures. The HPCS system also serves as a backup to the RCIC system. The system is designed to operate from normal offsite auxiliary power or from its dedicated emergency diesel generator.

The HPCS system is required to start and operate reliably over its design operating range. During the LOFW event and isolation transients, the RCIC maintains water level above the top-

Section 4.3 of the CPPU LTR) is based on the current LPCS capability, and will confirm on a plant-specific basis that the system provides adequate core cooling. The staff further reviewed ECCS system performance, as discussed in Section 4.3 of this SE. The staff finds the proposed evaluation and confirmation approach acceptable.

The CS/LPCS system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS/LPCS system is to provide reactor vessel coolant inventory makeup for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. It also provides spray cooling for long-term core cooling in the event of a LOCA. The adequacy of the CS/LPCS system performance is discussed in Section 4.3 of the CPPU LTR. There is no expected change in the reactor pressure at which the CS/LPCS is required. [

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The staff agrees that this section [

] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

4.2.4 Low Pressure Coolant Injection

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA and, in conjunction with other ECCS systems, the LPCI mode is used to provide adequate core cooling for all LOCA events. The licensee will confirm that the existing system has the capability to perform the design injection function of the LPCI mode for operation at the CPPU condition and that the generic evaluation in Section 4.1 of ELTR2 [

.] Since the ECCS-LOCA analysis (see Section 4.3 of the CPPU LTR), based on the current LPCI capability will demonstrate that the system provides adequate core cooling, the staff finds the proposed approach acceptable.

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. There is no change in the reactor pressures at which the LPCI mode of RHR is required. The primary purpose of the LPCI system is to help maintain reactor vessel coolant inventory for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. The adequacy of this system is discussed in Section 4.3 of the CPPU LTR.

The staff agrees that this section [

] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

4.2.5 Automatic Depressurization System

The ADS uses relief or safety/relief valves (SRVs) to reduce reactor pressure after a small-break LOCA, allowing the LPCI and CS/LPCS systems to provide cooling flow to the vessel The adequacy of this system is discussed in Section 4.3 of the CPPU LTR. CPPU does not change the conditions at which the ADS must function. The plant design requires the SRVs to have a minimum flow capacity. After a specified delay, the ADS actuates either on low water level plus high drywell pressure or on sustained low water level alone. The licensee will confirm that the ability of the ADS to initiate on appropriate signals [.] Since the licensee's ECCS-LOCA analysis (see Section 4.3 of the CPPU LTR), based on the current ADS capability, demonstrates that the system provides adequate core cooling, the staff finds the evaluation acceptable.

The staff agrees that this section [

] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

4.2.6 Emergency Core Cooling System Net Positive Suction Head

Operation at CPPU conditions increases the reactor decay heat, which increases the heat addition to the suppression pool following a LOCA event. As a result, the long-term peak suppression pool water temperature and long-term peak containment pressure may increase. The most limiting case for NPSH typically occurs at the peak long-term suppression pool temperature. The ECCS NPSH was evaluated in Section 4.1.8.5 of ELTR2, Supplement 1, Volume I. For HPCI, HPCS, CS/LPCS and RHR/LPCI systems, changes in the peak long-term suppression pool temperature and containment pressure are determined by the containment analyses (Section 4.1 of the CPPU LTR). If these values are bounded by the previous evaluation, no additional plant-specific analyses are required for the NPSH.

[

.] The CPPU LTR states that the ECCS NPSH evaluation will be based on the methodology described in Section 4.1.8.5 of ELTR2. This approach involves a plant-specific analysis of the effect of the increased wetwell temperature on NPSH. To the extent credited in the current design basis, the approach credits positive containment pressure to augment NPSH. The staff finds this approach acceptable. However, if, due to the effects of power uprate, this positive containment pressure is credited for a longer duration or a higher magnitude, then these changes would be subject to additional review.

4.3 Emergency Core Cooling System Performance

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

The CPPU approach takes an exception to the guidelines given in ELTR1. The ELTR1 approach called for a complete plant-specific break spectrum evaluation to be submitted as part of the PUSAR, using equilibrium core design parameters. In the CPPU approach, the LOCA analysis description is based on [

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The CPPU approach [ ] is judged to be acceptable for the following reasons:

[

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In addition to imposing these restrictions, GENE justified the application of this methodology to the following seven instrument setpoints·

[

]

GENE has justified each of these instrument setpoints and allowable values based on the fact that either the CPPU has no effect on instrumentation error or is not credited in the accident analysis, or the magnitude of the error has no effect on the analysis. GENE provided enough basis to demonstrate for each of these instrument setpoints that the simplified method will not have any effect on the plant's licensing basis [

.] Therefore, the staff finds the simplified instrument setpoint methodology discussed in the LTR for extended power uprate acceptable.

5.3 BOP Monitoring and Control

No safety-related setpoint change for these systems is required as a result of the uprate, with the exception of main steam line high flow. Main steam line high flow is discussed in Section 5.3.1 of the CPPU LTR and is considered in Section 10.1.1 of the staff's SE.

5.4 Technical Specification Instrument Setpoints

TS instrument allowable values and/or setpoints are those sensed variables that initiate protective actions and are generally associated with the safety analysis. The determination of instrument allowable values and setpoints generally includes consideration of measurement uncertainties and is derived from the conservative analytical limits used in specific licensing or safety evaluations. Increases resulting from CPPU in the core thermal power and steam flow affect some instrument setpoints. The following setpoints are discussed in this section:

- APRM Flow-Biased Scram
- Rod Worth Minimizer/RCIS Rod Pattern Controller Low Power Setpoint
- Rod Block Monitor
- RCIS Rod Withdrawal Limiter High Power Setpoint
- APRM Setdown in Startup Mode

5.4.1 <u>APRM Flow-Biased Scram</u>

[

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[

] The TS will be modified by adjusting the flow-biased scram setpoint.

5.4.2 Rod Worth Minimizer/RCIS Rod Pattern Controller Low Power Setpoint

he rod worth minimizer/RCIS rod pattern controller low power setpoint is used to bypass the rod pattern constraints established for the control rod drop accident at low power levels. [

]

5.4.3 Rod Block Monitor

The severity of rod withdrawal error during power operation event is dependent upon the RBM rod block setpoint. [

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5.4.4 RCIS Rod Withdrawal Limiter (RWL) High Power Setpoint (HPSP)

[

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5.4.5 APRM Setdown in Startup Mode

The value for the TS safety limit for reduced pressure or low core flow conditions may be reduced to satisfy the fuel thermal monitoring requirements established as described in Section 2.1 of the CPPU LTR. The setpoint for the APRM setdown in the startup mode is based on the TS setpoint. The current TS may be based on either a conservative generic setpoint or on a plant-specific calculated value.

5.5 Conclusion

GENE has justified each of these instrument setpoints and allowable values based on the fact that either the CPPU has no effect on instrumentation error or is not credited in the accident analysis or the amount of error has no effect on the analysis. The staff reviewed the CPPU LTR discussion and finds the simplified setpoint methodology for these instruments under the conditions specified in the LTR acceptable. This is based on the staff's expectation that licensees referencing the CPPU LTR will justify any plant-specific differences from the CPPU LTR with respect to instrumentation setpoint methodologies.

Based on the above review and evaluation of the LTR and GENE's responses to the staff's RAI, the staff concludes that instrument setpoint changes for CPPU are acceptable.

the AC power system is to be performed for CPPU to assure an adequate AC power supply to safety-related systems. The staff finds this to be acceptable.

6.2 Direct Current (DC) Power

GENE stated that experience with previous power uprates has shown that the DC loads are not significantly increased because of power uprate. System loads are computed based on equipment nameplate data. [ ], the DC power distribution system is adequate for CPPU.

The staff concluded that [ ], the design for DC power systems is acceptable for CPPU.

6.3 Standby Liquid Control System

The standby liquid control system (SLC) is a manually operated system that pumps concentrated sodium pentaborate solution into the reactor vessel in order to provide neutron absorption. It is designed to be capable of bringing the reactor to a subcritical shutdown condition from rated thermal power.

An increase in the core thermal power does not by itself directly affect the ability of the SLC boron solution to bring the reactor subcritical and to maintain the reactor in a safe-shutdown condition. A higher fuel batch fraction, a change in fuel enrichment, or a new fuel design may affect the shutdown concentration, but operating at the CPPU condition does not affect the required boron solution. The SLC system shutdown capability is reevaluated [ ]. The effect of the CPPU on the SLC system injection and shutdown capability will be evaluated [.]

The SLC system is designed to inject at a maximum reactor pressure equal to the upper analytical setpoints for the lowest group of SRVs operating in the relief mode Since the reactor dome pressure [ ] will not change, [.] The SLC pumps are positive displacement pumps, and small changes in the SRV setpoint would have no effect on the SLC system capability to inject the required flow rate. The licensee will confirm [ ] that there is sufficient margin to lifting the SLC system relief valves. The calculated maximum required pump discharge pressure, based on the peak reactor pressure during the limiting ATWS event, should be below the lowest calculated nominal opening pressure for the SLC pump relief valves. Consequently, the SLC relief valves would not lift during the ATWS events. The operation of the SLC system is also analyzed to confirm that the pump discharge relief valves will reclose in the event that the system is initiated before the time that the reactor pressure recovers from the first transient peak. The evaluation compares the calculated maximum reactor pressure needed for the pump discharge relief valves to reclose with the lower reactor pressure expected during the time the SRVs are cycling opened and closed. Considerations are also given to system flow, head losses for full injection, and cyclic pressure pulsations due to the positive displacement pump operation in determining the setpoint for the relief valve. The relief valves are periodically tested to maintain this tolerance. Otherwise, it is expected that the relief valves will operate as designed and originally tested.

neutron reactions as a secondary result of reactor power. The radiation sources in the core during operation are expected to increase in proportion to the increase in power. However, this increase is bounded by the existing safety margins of the design basis sources. Since the reactor vessel (inside the fully-inerted primary containment) is inaccessible during operation, a proportional increase in the radiation sources in the reactor core will have no effect on occupational worker personnel doses during power operations. Due to design shielding and containment surrounding the reactor vessel, worker occupational doses are largely unaffected, and doses to the public from radiation shine from the reactor vessel remain essentially zero as a result of the power uprates. Potential impacts of increased dose rates inside primary containment on component reliability are discussed in Section 10.3 of the staff's safety evaluation.

From a post-operation perspective, GENE discussed the two separate sets of radiation source data for the core, and both must be corrected for radioactive decay after shutdown. The first, the gamma-ray source, is used for radiation shielding calculations for the core and individual fuel bundles. In terms of MeV/sec per reactor thermal power, this source is a function of, and increases in proportion with, reactor power. The second set of post-operational source data is the nuclide activity (fission products primarily) in the fuel. This data is used as input for post-accident and spent fuel analyses, which apply appropriate regulatory modeling for source term release fraction, timing and transport assumptions and parameters. Both short-lived and long-lived nuclides are expected to increase in approximate proportion to increase in core thermal power. GENE discusses appropriate decay and equilibrium considerations, and establishes bounding parameters to be used for core radiation source calculations. [.] Plant power uprate applications that conform with the values of these bounding parameters would be acceptable.

However, as discussed in Section 8.5 of this SE, in order to follow NUREG-0737, Item II.B.2, post-accident shielding requirements, licensees would need to perform plant-specific analyses of post-accident dose rates as they affect operator access to designated vital areas.

8.4 Radiation Sources in the Reactor Coolant

Radiation sources in the reactor coolant contribute to the plant radiation levels. These sources include coolant activation products, activated corrosion products (ACP) and fission products. GENE examined the impact of the power uprate on each type of source. The staff accepts the approach described below to address CPPU effects on radiation sources in reactor coolant.

8.4.1 Coolant Activation Products

During operations, the reactor coolant passing through the reactor core region becomes radioactive as a result of nuclear reactions. GENE notes that the activation product concentrations in the steam [ ] following the power uprate since the increase in activation production in the steam passing through the core is [ ] with the power increase, but [ ] by the increase in steam flow through the core. [[ ]] the transit time from the core to the turbine building components will be reduced (due to increased steam flow rate). This decrease in transit time reduces the decay period of very short-lived radionuclides (mainly N–16), resulting in higher dose rates, roughly proportional to the power increase, in and around the turbine/condenser and other main steam components.

Because of plant-specific design and varying operational chemistry regimes, the percent increase in activation products (and operational doses rates) as a result of the power uprate will be determined [.]

8.4.2 Activated Corrosion Products

ACPs result from the activation of metallic corrosion and wear materials in the reactor coolant, and are expected to increase as a result of a power uprate. The equilibrium level of ACPs in the reactor coolant is expected to increase as a result of the increase in feedwater flow rate and the increase in neutron flux in the reactor. The increased feedwater flow will likely reduce the efficiency of the condensate filtration and demineralization system (CFDS), thereby resulting in an additional increase in the equilibrium level of ACPs (and increased external dose rates). However, GENE expects that the ACP increase will not exceed the design basis concentrations.

Because of plant-specific design of the CFDS and feedwater systems, and varying operational chemistry regimes, the increase in ACP as a result of the power uprate will be determined by a plant-specific analysis.

8 4.3 Fission Products

Fission products in the reactor coolant result from the escape of minute fractions of the fission products in the fuel rods. Fission product release into the primary coolant is dependent on the nature and number of fuel defects and GENE does not expect an increase in these defects as a result of the power increase. [

.] Given that current levels of fission product activity typically found in reactor coolant and steam are [ ], a percent fission product increase of no more than the power uprate [ ]

Because of potential plant design and operational differences, [

.]

8.5 Radiation Levels

External radiation levels contribute to the plant worker occupational doses during plant operation, post operations (plant shutdowns), and during postulated accident conditions. These plant radiation levels result from activation and fission products, and ACP discussed in Section 8.4. GENE examined the impact of power uprates for each operational mode or condition.

GENE stated that many aspects/areas of the plant were conservatively designed for higher-than-expected radiation sources. Therefore, for most plants, the increases in radiation levels during operations at higher power levels will not affect radiation zoning or shielding adequacy for most plant areas. [

than the percentage increase in reactor thermal power. Given that the installed MOS in operating plants effectively reduces main offgas effluents by factors greater than 100, effluent release increases of up to 20 percent from the MOS are expected to have a negligible impact on calculated doses to the public. GENE concludes that the actual estimated increase in off-site doses from the MOS will be determined by a [ ] analysis to ensure that the public doses remain below the limits of 10 CFR Part 20, 10 CFR Part 50, Appendix I and 40 CFR Part 190.

Gamma radiation (skyshine) from coolant activation products (chiefly Nitrogen-16) in the reactor steam in the main steam system components in the turbine building provides another offsite public dose pathway. GENE notes that the power uprate results in increased steam flow, leading to generally proportional higher levels of activation products (chiefly Nitrogen-16) and resultant external dose rates in and around the turbine building. Typical shielding design more than adequately bounds any such radiation level increase due to power uprate.
During power operations, N-16 production is increased by the HWC process (routine hydrogen gas injection into the reactor feedwater in an effort to prevent intergranular stress corrosion cracking of reactor internals). The resulting higher dose rates then increase the gamma skyshine both on- and off-site. Applicants should be aware of the impact on station workers working in buildings adjacent to the turbine building (e.g., administrative station employees that may be designated as members of the public). These station employees then would be subject to the 10 CFR Part 20 public dose limits. For plants that use HWC, a site-specific analysis will need to be performed to confirm that the turbine building skyshine increases due to power uprate do not result in doses to members of the public exceeding the limits in 40 CFR Part 190.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

This section addresses the evaluations in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Chapter 15, that are documented in the current plant power uprate submittals. These reactor safety performance evaluations include:

- Anticipated Operational Occurrences
- Design Basis Accidents
- ATWS
- Station Blackout

Plant-specific evaluations will be included in the plant-specific submittal consistent with the format and level of detail of previous extended power uprate submittals or as discussed in the CPPU LTR sections. The applicability of the generic assessments for a specific plant application will be evaluated. The plant-specific submittal will either document the successful confirmation of the generic assessment or provide a plant-specific evaluation if the applicability assessment is unsuccessful.

The staff agrees that this section [

] the plant must perform a [ ] evaluation using an approved methodology. This approach is acceptable to the staff.

.] This is acceptable to the staff.

All of the transients listed in Table E-1 of ELTR1 were considered. The limiting overpressure transient will be analyzed as defined by GESTAR-II and by the ELTR1. [

.] Analyses of the events listed in Table E-1 of ELTR1, for plants pursuing extended power uprates, have confirmed the applicability of the GESTAR-II list of limiting events. [

.]

The limiting events are defined in GESTAR-II and the core reload analysis will be based on approved GESTAR-II methodology. The other events listed in Table E-1 of ELTR1 do not establish the OLMCPR, based on experience and the characteristics of these events, and therefore are not analyzed to establish this limit.

As discussed above, most of the transients listed in Appendix E of ELTR1 will be analyzed [

] these evaluations are not expected to be included in the power uprate license amendment submittal. [

]. [

] The results of the limiting thermal margin