GE Nuclear Energy

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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Tae Kim, Lead Project Manager
Project Directorate III-1

SUBJECT: **NRC Request for Additional Information (RAIs) on GE Licensing Topical Report NEDC-32424P, "Generic Guidelines for GE Boiling Water Reactor Extended Power Uprate", (TAC No. M91680)**

REFERENCE: NRC letter to GE (Mr. David J. Robare) dated September 28, 1995, "Staff Comments and Questions Regarding GE Licensing Topical Report NEDC-32424P.

Enclosed are GE responses to the above-referenced RAIs #1-12 and 14. Response to RAI #13 will be provided in a separate, proprietary submittal.

Please let me know if you have any questions or comments concerning these responses.

Sincerely,

David J. Robare

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Question 1:

It is stated in the ELTR1 that the generic guidelines contained in the ELTR1 are based in Part on one complete uprate analysis to 120% of the original rating. Please provide from that analysis:*

a) All NSSS (nuclear steam supply system modifications (both minor and major) that were identified as necessary;

b) All BOP [balance-of-plant] modifications (both minor and major) that were identified as necessary; and

c) Identified changes to the technical specifications

Response 1:

a) Nuclear steam supply system modifications
CRD Hydraulic: Refurbish one CRD pump to original design specifications
Miscellaneous NSSS: setpoint changes, revised Technical Specifications (see 1c below)

b) BOP [balance-of-plant] modifications -

Turbine: New rotors (High and Low pressure turbines),New Generator rotor, Upgrade generator cooling, New turbine control and bypass valves.
Isophase Bus: Improved cooling
Main Transformer: Cooler upgrade
Condensate Pumps: New Impellers
Heater Drain Pumps: New #5 Heater drain Pumps
Feedwater Pumps: New Impellers
Miscellaneous BOP setpoint changes

c) A review of plant Technical Specifications has determined that the areas indicated below will require updating for power uprate. All the values provided are specific to that plant evaluated and most will be different for other plants. For purposes of being consistent with the Technical Specifications, all pressure values in bars are above atmospheric.

Some of the setpoint changes identified in this Section are dependent on the operating power flow map that is used when Extended Power Uprate is implemented. In particular, changes in the minimum allowable core flow at uprated power, the maximum dome pressure, or in the power level itself, will affect some of the setpoints. The values reported below are based on implementing a power uprate to 3600 MWt, with a corresponding increase in dome pressure of 1.38 bar (20 psi). Section and Item numbers refer to the particular plants Tech Specs.

SECTION 1.0 DEFINITIONS

Item AG Rated Thermal Power

The rated reactor power definition will require revision each time the power level is increased, up to a maximum of 3600 MWt

SECTION 2.1 SAFETY LIMITS

Item B Thermal Power High Pressure and High Flow

The safety limit Minimum Critical Power Ratio (MCPR) of 1.07 may require revision as the power level is increased.

SECTION 2.2 LIMITING SAFETY SYSTEM SETTINGS

Item A Reactor Protection System Instrumentation Setpoints

1. Average Power Range Monitor

   Slope for Flow Biased nominal trip setpoint (NTSP) and allowable value is rescaled to account for change to rated power, such that absolute value of setpoint (in MW and kg/sec) is unchanged. (Note that this particular plant implemented a reduction in minimum core flow at rated power which is not be part of the generic uprate process of ELTR1, because no change in flow to the maximum flow control line of above the presently licensing MELLA line is requested. That reduction in core flow necessitated a further change in the setpoint)

2. Reactor Vessel Steam Dome Pressure High

   The setpoint should be increased to 76.2 bar (1105 psig) for the NTSP and 77.2 bar (1120 psig) for the allowable value.

Item B Isolation Actuation Instrumentation Setpoints

1. Main Steam Line Isolation Flow High

   The setpoint in bar corresponding to 140% of rated will require recalibration at uprated power.

2. Reactor Core Isolation Cooling System Isolation RCIC Steam Line Flow High

   The acceptability of the current setpoint will have to be determined during startup at uprated power.

Item D ATWS Recirculation Pump Trip and ARI System Instrumentation Setpoints

1. Reactor Vessel Pressure High

The setpoint should be increased to 76.5 bar (1110 psig) for the NTSP and 77.2 bar (1120 psig) for the allowable value.

Item H Control Rod Block Instrumentation Setpoints

1. Average Power Range Monitor

Slope for Flow Biased nominal trip setpoint (NTSP) and allowable value is rescaled to account for change to rated power, such that absolute value of setpoint (in MW and kg/sec) is unchanged. (Note that this particular plant implemented a reduction in minimum core flow at rated power which is not be part of the generic uprate process of ELTR1, because no change in flow to the maximum flow control line of above the presently licensing MELLA line is requested. That reduction in core flow necessitated a further change in the setpoint)

SECTION 3.1/4.1 REACTIVITY CONTROL SYSTEM

Item D Control Rods Maximum Scram Insertion Times

1. Pressure values for CRD scram time testing

The pressure values for the CRD scram time testing should be increased from 65.5 bar (950 psig) to 66.9 bar (970 psig) and from 72.4 bar (1050 psig) to 73.8 bar (1070 psig).

Item I Standby Liquid Control System (SLCS)

1. Minimum sodium pentaborate concentration (equivalent weight)

The value of 2280 kg (5027 lb.) may change on a cycle specific basis.

2. Pump relief valve setpoint

The value of less than or equal to 97 bar (1407 psig) should be modified to read "greater than or equal to 93.6 bar (1358 psig) and less than or equal to 99.4 bar (1442 psig)".

SECTION 3.2/4.2 POWER DISTRIBUTION LIMITS

Average Planar Linear Heat Generation Rate (ALHGR)

1. Foot Note related to maximum core flow.

Foot Note should be modified to allow a maximum flow of 106% at rated feedwater temperature and 108% with 28°C (50°F) or more of feedwater temperature reduction. (Note that this particular plant implemented a change to the maximum core flow (a different value for increased core flow) which would not necessarily be required for all extended uprates.)

2. Flow Dependent MAPLHGR Limit

   Curve should be included to allow operation at up to 108% core flow.

3. Power Dependent MAPLHGR Limit

   Current limits above 40% power are applicable to uprated power operation. The flow dependency in the limits below 40% power can be eliminated for GE10 fuel.

Item B Minimum Critical Power Ratio (MCPR)

1. Maximum core flow at EOC

   The maximum flow at end of cycle conditions should be modified to allow a maximum flow of at least 108% of rated.

2. Flow biased MCPR figures for various fuel types (MCPRf curves)

   Figures should be modified to provide limits for a maximum flow of up to 108% of rated.

3. Power biased MCPR figures for various fuel types (MCPRp curves)

   Current limits above 40% power are applicable to uprated power operation. For GE10 fuel at or below 40% power, the limits may be relaxed to a constant limit of 1.80. A similar relaxation should be possible for SVEA 96 fuel.

SECTION 3.4/4.4 REACTOR COOLANT SYSTEM

Item A Recirculation System Recirculation Loop

1. Maximum core flow

   The maximum core flow should be increased from 105% to 106% with normal feedwater temperature and 108% with 28°C (50°F) or more of feedwater temperature reduction.

2. Power/Flow map

   The power/flow map figures need to be updated to be consistent with the analyzed operating map shown in Section 2.

Item E Safety Relief Valves (SRV)

1. SRV setpoints

   The SRV spring and relief mode setpoints are increased. (Vessel operating pressure was increased.)

Item F   Safety Relief Valves (SRV)   Low Low Set

1. Low low set SRV function setpoints

   The low low set SRV function setpoints are increased.

Item K   Reactor Vessel Toughness Surveillance Program

1. Reactor Vessel Pressure and Temperature Limits.

   These figures could be updated to account for the lower calculated neutron fluence resulting from placing highly depleted fuel in the peripheral locations. Leaving them unchanged would be conservative. (Note that the generic power uprate guidelines in ELTR1 reevaluate these curves and may cause them to shift to higher temperatures if the vessel wall fluence is increased.)

Question 2:

Please provide the proposed outline of the ELTR2 and a brief description of each generic evaluation and analysis that will be covered in the ELTR2.

Response 2:

The generic evaluations are planned in two submittals, one in December 1995 and one in May 1996. All topics addressed in LTR2 will be addressed in the two submittals. The outline and descriptions are grouped according to submittal date:

Contents of ELTR2, Generic Evaluations of GE BWR Extended Power Uprate December 1995:

2. LICENSING EVALUATIONS

   2.2 Setpoint Methodology

   Description of setpoint methodology application to extended uprate, similar to LTR1, (stretch (5%) uprate program)

   2.3 Emergency Operating Procedures

   List (similar to LTR1) of EOP parameters which must be reevaluated as part of an extended uprate. Basis will be more recent EPG version than LTR2.

   2.4 Probabilistic Safety Assessment (performed by Northern States Power, Monticello, BWR/3, 12% uprate)

   The impact of power uprate on plant risk will be assessed by reviewing the Individual Plant Examinations (IPE). The assessment of the effect of power uprate on the plant IPE will consider the effect of power uprate on IPE inputs and assumptions such as: Initiating Event Frequency, Success Criteria, Component Failure Rates, Time Available for Operator Action and Equipment Restoration. As part of the IPE, the utilities were required to identify any plant vulnerabilities associated with core damage potential and containment performance. The scope of the study will be sufficient to identify if any new vulnerabilities are introduced by the power uprate. If any new vulnerabilities are identified they will be reported in the Licensing Report following the guidelines of GL 88-20.

3. ANALYTICAL EVALUATIONS

   The following sections provide generic evaluations of aspects of power uprate that involve analytical investigation. In some areas, cases are presented which bound specific plant sizes and/or BWR product lines. In others, a generic review of the impact of power uprate is provided to show that performance of all applicable plants remains within acceptance criteria or current licensing practice. In this way, generic review will significantly help the review of the individual lead projects and subsequent applicants.

3.1 Loss of Feedwater Flow Transient (Bounding* BWR/3)

This section documents the generic basis and results for evaluation of the Loss of Feedwater Flow transient event. This case is the original design basis for the performance of the Reactor Core Isolation Cooling (RCIC) System in BWR/3 plants. The RCIC System is designed to maintain adequate water level in the reactor during a Loss of Normal Feedwater transient even with single failure of the other high pressure water supply system [High Pressure Coolant Injection (HPCI)]. The RCIC System should maintain sufficient water level inside the core shroud to assure that the top of the active fuel remains covered throughout the event. In principle, the other high pressure system must also meet the same performance requirements with the assumed single failure of the RCIC System, but that is always less limiting, since RCIC is the smaller of the two systems on all plants.*

In response to an NRC request that the analysis bound applications to BWR/3 plants, parametric studies will be included to show the variation from the typical BWR/3 response, through a range of key parameters which bound the product line. (e.g. RCIC capacity, vessel size/power). Also, the plant configuration assumed in the analysis will be documented and a "roadmap for application to subsequent BWR/3 submittals" will describe the process applied to determine whether a plant is bounded by the analysis.

3.2 Stability

Justification that the current, NRC approved BWROG Long term stability solutions are applicable to extended power uprate. Constraints on possible expansion of the high-power/low-core-flow portion of the operating range are considered in the power uprate analysis and in defining the plant startup procedures. Operation will not be expanded beyond the region previously licensed for the applicable product line. Appendix C of ELTR1 defines the specific guidelines to be followed so that power uprate plants remain within the previously licensed range of core flow at the current rated power for each GE BWR product line. These guidelines will be used for selecting the operating range for power uprate. The specific ranges previously licensed for each product line will be maintained so that the power uprate will have no detrimental effect with regard to stability. In this way, the uprate will maintain stability protection at the same level as agreed upon for non-uprated operation of the applicable product line. Since the exclusion region boundaries are redefined such that the absolute powers and core flows are the same as the current boundaries, power uprate will not affect stability.

3.3 Core Spray Distribution

Justification (similar to LTR2) of the applicability of core spray distribution analysis assumptions at power uprate conditions based on GE LOCA/ECCS methods.

3.4 Safety Limit Minimum Critical Power Ratio (SLMCPR)

Description of the plant specific process applied to determine the SLMCPR applicable after an extended uprate, such that the original basis (avoidance probability of rods entering transition boiling) is unchanged. Example evaluation of SLMCPR for a 20% uprate.

3.6 Materials and Coolant Chemistry

Justification that the constraints which address Intergranular Stress Corrosion Cracking and Erosion/Corrosion are sufficient for operation at the uprated power level. Evaluation of the effect of extended uprate on coolant chemistry.

3.7 ATWS Evaluations (Bounding BWR/3)

ATWS evaluation of a BWR/3 per Appendix L of ELTR1 to show the effect of a 20% uprate, relative to compliance with the NRCs rule on ATWS. ATWS rule compliance primarily involves alternate shutdown equipment which has been previously installed at each unit. The equipment will remain and its performance at any changed conditions (due to uprate) will be evaluated (e.g., higher steam flow). Power uprate operation does not significantly affect the long-term ATWS response because it does not involve a uniquely higher rod line, and, therefore, there is no increase in the power level following the ATWS recirculation pump trip.*

In response to an NRC request that the analysis bound applications to BWR/3 plants, parametric studies will be included to show the variation from the typical BWR/3 response, through a range of key parameters which bound the product line. (e.g. SRV capacity, Initial pressurization rate). Also the plant configuration assumed in the analysis will be documented and a "roadmap for application to subsequent BWR/3 submittals" will describe the process applied to determine whether a plant is bounded by the analysis.

3.8 ASME Over-Pressure Protection Evaluations (BWR/3,4,5&6)

Evaluation of the effect of 20% uprate on the ASME over-pressure protection requirements on a BWR/3,4,5&6 plant, including sensitivity to SRVs out of service (reducing SRV capacity) and initial dome pressure.

5. IMPACT ON SAFETY MARGIN

5.1 Fuel Thermal Limits
5.2 Design Basis Accidents
5.3 Transient Evaluations
5.4 Environmental Consequences
5.5 Technical Specification Changes
5.6 Conclusion

Section 5. will describe the effect of extended uprate on safety margin in each of the above areas. The process applied will be that presented in the April 1995 NRC meeting on extended power uprate, "No significant hazards".

Contents of ELTR2, Generic Evaluations of GE BWR Extended Power Uprate Supplement 1, May 1996:

2. LICENSING EVALUATIONS

2.1 Generic Communications

Identification of Generic communications which must be specifically addressed in plant specific licensing report, those which are generically addressed by the power uprate analysis process described in ELTR1, and those which are not affected by power uprate. The same process which has been described to the NRC staff and approved in developing the lists in ELTR1 will be applied.

3. ANALYTICAL EVALUATIONS

3.5 Containment Atmosphere Combustibility

Evaluation (similar to ELTR2) of the plants compliance in a 20% uprate with 10CFR50.44 and 10CFR50.46

4. HARDWARE CAPABILITY EVALUATIONS

4.1 Low Pressure Emergency Core Cooling Systems (ECCS)
4.2 High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems
4.3 High Pressure Core Spray (HPCS)
4.4 Control Rod Drives and Scram Performance
4.5 Recirculation System
4.6 Safety Relief Valves
4.7 Main Steam Isolation Valves

For each system listed above, the impact due to power uprate is evaluated (similar to ELTR2) assuming the following primary operating condition changes:· Increased power level of 20% (i.e., heat flux, stored heat, fission products, neutron fluence)· Increased reactor pressure (1095 psia).· Increased reactor temperature (556°F).· Increased steam and feedwater flow rates of ~24%.

4.8 Generic Piping

The generic methodology developed by GE to evaluate the effect of power uprate on piping with be applied to determine the effect of a 20% uprate on BWR/5 piping systems. This methodology has been reviewed by NRC. as part of LTR2.

Question 3:

The evaluation of the reactor vessel and internals for the extended power uprate should consider the potential for increased dynamic loadings due to the annulus pressurization, jet thrust forces and the biological shield wall motions resulting from loss-of-coolant accident (LOCA) and safety/relief valve (SRV) discharge events.

Response 3:

The reactor vessel and internals will be evaluated according to the current licensing criteria as stated in Section B.2(2) of the extended power uprate licensing topical report (NEDC-32424P). The evaluation will consider the potential for increased dynamic loads due to the annulus pressurization and jet thrust forces resulting from LOCA within the annulus region, and other loads from LOCA and SRV discharge events if the current licensing basis includes such loads.

The biological shield wall motion will also be considered if it is included in the current plant-specific licensing basis.

Question 4:

The staff believes that GE should provide, to the fullest extent possible, appropriate bounding evaluations for the extended power uprate regarding the structural integrity of control rod drive mechanism and reactor internals. Will such evaluations be included in the ELTR2?

Response 4:

Evaluations for extended power uprate regarding structural integrity of reactor internals will be performed as part of plant-specific uprate analyses, because the loadings on internals are plant-specific.

Evaluations for extended power uprate regarding structural integrity of control rod drive mechanisms will be performed and included in Section 4.4 of ELTR2.

Question 5:

The structural integrity of the reactor vessel material should be evaluated with the uprated conditions in accordance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99, Revision 2, 'Radiation Embrittlement of Reactor Vessel Materials," Specifically:

a) Neutron fluence calculations should be re-evaluated, and the evaluation should provide neutron fluence, neutron fluence spectrum, and flux at the end of license;

b) Pressure-temperature limits in the technical specifications should be re-evaluated and the evaluation should provide the limiting adjusted reference temperature in accordance with Regulatory Guide 1.99, Revision 2. The evaluation should also provide neutron fluence at the applicable effective full power years of the pressure-temperature limits;

Response 5:

a) This will be done in plant specific uprate submittals. However, when the uprate is applied for, the exact core configuration is not yet determined. Therefore, an approximation of the change in fluence is made for the purposes of Appendix G and Regulatory Guide 1.9? calculations of uprate impact. The approximation followed is to increase the fluence by the percentage of the uprate.

b) This will be done in plant-specific uprate submittals, using the approximate fluence discussed in 5a.

Question 6:

The reactor vessel material surveillance program should be evaluated for the uprated conditions in accordance with Appendix N to 10 CFR Part SO. Specifically, with the increased neutron fluence, the surveillance capsules withdrawal schedule and number of capsules should be re-evaluated to determine if the schedule and the number of capsules Satisfy Appendix H to 10 CFR Part 50.

Response 6:

Based on past power uprate analyses, uprate typically increases adjusted reference temperatures less than 10°F over the life of the plant. This is a small change in the overall vessel embrittlement, so there is no need to consider any change to existing BWR surveillance programs.

Question 7:

The structural integrity of the core shroud should be re-evaluated with uprated conditions, such as increased operating pressure and flow rate. Also, the crack growth model should be re-evaluated considering the potential effect of the irradiation assisted stress corrosion cracking (IASCC).

Response 7:

Structural Integrity of the Shroud for Uprated Power Conditions

The stresses in the shroud due to the core ΔP are very small, less than 2 ksi for 30 psi pressure (below the core plate). Increase in ΔP due to power uprate/increased core flow will not result in a significant increase in stress in the shroud. The limiting condition for the shroud is the seismic event, which remains unchanged.

Potential Increase in IASCC susceptibility

Even in the absence of Power Uprate, the fluence in the shroud will eventually be higher than the threshold of IASCC. Extended power uprate could result in the threshold being reached somewhat earlier in time. Given that the IASCC mechanism is active, the slightly higher fluence will not lead to any drastic increase in the crack growth rate. IASCC mitigation measures (such as moderate hydrogen water chemistry, noble metals) are applicable under extended power uprate just as they are under the rated power conditions.

Question 8:

Would the change in fluid velocity, temperature and moisture content of two phase fluid as a result of extended power uprate lead to damages to carbon steel components by flow accelerated corrosion (FAC)? The NRC in Generic Letter 89-08, 'Erosion/Corrosion - Induced Pipe Wall Thinning, requested all the licensees to implement a long-term monitoring program based on the EPRIs CHECWORKS computer code. The computer code predicts potential damage to the carbon steel components caused by FAC and permits the licensee to identify and repair or replace defective components before their failure occurs. The program is inspected by the NRC inspectors. Any increase in FAC which may occur due to power uprate should be addressed.

Response 8:

An increase in fluid velocity would be detrimental to carbon steel as a strong function of the dissolved oxygen content. The amount of increase in FAC will be evaluated on a plant-specific basis using CHECWORKS or an alternate methodology suitable for addressing the Generic Letter 89-08 erosion/corrosion concern. This methodology determines inspection intervals and locations to preserve the integrity of carbon steel components.

Question 9:

The piping evaluations for extended power uprate should include feedwater and other impacted safety-related BOP piping. The effects of increased structure/building motions, resulting from LOCA and SRV discharge events and subcompartment pressurization, on the piping and system components should be addressed.

Response 9:

One of the first tasks addressed by GE when performing a power uprate program is to review the P&IDs of all safety-related systems that are impacted by power uprate. The purpose of these reviews, conducted with the GE Lead Systems Engineers (LSEs), is to identify all piping that is impacted by power uprate design parameter increases. As a result of this review, the Feedwater and other safety-related piping that are impacted by power uprate are identified and, therefore, are included in GEs piping power uprate evaluation process.

The plant-specific evaluation includes an assessment of the impact of power uprate design parameter changes on the existing piping analysis basis including load definition. An assessment is made of the impact to LOCA, SRV discharge and subcompartment pressurization. If a load definition is impacted, the effect of the change on piping and systems components due to 'increased structure/ building motion is evaluated.

Question 10:

The high-energy line breaks, including break locations and subsequent effects (jet impingement and pipe whip) should be evaluated since the pipe stresses, rupture thrust forces. and gap settings between pipe and restraints may increase as a result of the extended power uprate.

Response 10:

GEs plant-specific piping power uprate evaluations include an assessment of high-energy line break locations to assure that no new breaks locations are postulated.

Gap settings (C2,C3) between the pipe and the pipe whip restraint are reviewed for power uprate impact to assure the proper positioning of the restraint with respect to the centerline of the pipe after heatup to normal operation.

GE also provides jet impingement and pipe whip evaluations which predict the increase in jet impingement loading with respect to the jet centerline. The increase in loading on pipe whip restraints is determined in order to assure that the pipe whip restraint has adequate design margin to accommodate the 'increase in pipe whip loading.

Question 11:

The overall structural integrity of the turbine should be re-evaluated in terms of vibration analysis for the increased steam flow. In particular, the vibration response of low pressure turbine blades caused by the increased steam flow should be evaluated.

Response 11:

All stages of the turbine will be screened for excessive loadings and/or stress to insure that increased steam flow will not negatively affect the integrity of the long blades in the last few stages of the low pressure turbine.

A mechanical review of the rotors will be conducted to evaluate steady-state vibrational and upset stress conditions affected by uprated steam conditions and to insure that turbine vibrations are acceptable under increased steam flow.

Question 12:

Section L.2.4. Provide the basis for determining when the large disturbance tests need to be part of the startup test program. What is the basis for choosing power uprate increases of 10% and 15% for the MSIV [main steam isolation valve] closure and generator load rejection tests, respectively.

Response 12:

The purpose of the initial startup large disturbance tests is to check that the plant components and equipment perform as designed. In the case of uprate, the plant has accumulated several years of experience in addition to the initial startup tests to show that the integrated plant performance is as designed.

After a significant uprate, i.e. 10%, it is prudent to repeat some of the initial startup tests. In selecting which tests to repeat it some considerations are:

1. Tests which will show that the integrated plant response is as expected.

2. Minimizing effect on plant capacity factor.

3. Eliminating tests where the there is little uncertainty in the result, or where similar data is available through normal operation, e.g. planned evolutions.

The large perturbation tests typically conducted in the initial startup are:

MSIV Isolation (All valves)
Turbine Trip
Single Recirculation Pump Trip (one of two pumps)
Single Feedwater Pump Trip (one of typically two pumps)

The MSIV isolation was selected when such data was not available at power levels with 10%* of the uprate power level, and the Turbine Trip when such data was not available at power levels with 15%* of the uprate power level.

The Single Recirculation pump trip was not selected, for inclusion since it not as large a disturbance. Previous tests have shown the plant to have large margins to the point where water level swell might trip the turbine. The margins are not expected to decrease enough at uprated power to affect the plants response. The normal startup evolution provides data near the power/ flow operating point at which this test stabilizes.

The Single Feedwater pump test response is not affected by the power increase. It is affected by the flow control line, but uprate utilizes the same MELLLA flow control that plants operate at already. The uprate will consider the effect of this event response in the Setpoint methodology (which can determine whether there is a significant change in the scram avoidance probability) and the review of the IPE (where the trip avoidance probability change, if any, would be considered). The scram probability can be reduced by further optimization of feedwater runout blocks and recirc runback settings. Some utilities choose to conduct an end of cycle single feedwater pump test to provide information on the plant response for operator training, and it's expected they will continue to do so, but no test is required in the initial uprate startup.

*In the case where the plant is implementing uprate in small increments, the provision to avoid the test if data is available from a previous test or an operating transient obviates the need for a special test. Many plants now have data acquisition systems which will automatically record inadvertent events during operation. Thus if the plant uprated in increments, the data might be available near the uprate power level from records of inadvertent events.

Question 14:

It appears that increasing reactor power will shorten the operating cycle length. As a result, licensees may wish to combine uprate with a cycle extension technique, such as final feedwater temperature reduction (FFTR) or increased core flow. Please discuss the potential consequences of cycle-extension combined with increased power level on fuel, including the impact on thermal limits and changes to end-of-cycle reactivity response. Additionally, please describe any changes to current fuel cycle management techniques which may occur as a result of an extended power uprate without FFTR or increased core flow.

Response 14:

If the cycle energy is not increased, the operating cycle length in days will be decreased. This affect is due to the increase energy output per unit time with the uprated power level. If the utility chooses to compensate for this change, the cycle energy can be increased by increasing bundle enrichment and/or increasing the number of bundles loaded in each cycle. In either case, the core design would continue to conform to GEs design and operating margin requirements.

End-of-cycle reactivity responses are driven more by the number of gadolinia rods and their concentration than by the changing power density. Again, in order to achieve loadings which conform to GE design requirements, the reactivity response at end of cycle would be very similar to a non-uprated core.

The APLHGR and LHGR thermal limits should not be affected by uprate. The CPR operating limit may increase, due to an increase in the safety limit, by less than 0.02, because of a flatter power distribution. The increased CPR safety limit allows the same margin to transition boiling to be maintained. The core design will provide design margin between the limits and actual parameters to provide maneuvering room for startups/shutdowns/rod sequence exchanges etc.

Final Feedwater Temperature Reduction (FFTR) and Increased Core Flow do not pose any problems combined with power uprate. The transient response in these operating modes will be determined as part of the power uprate engineering analysis, but the results should not be significantly more severe.

During operation, fuel cycle management techniques are unaffected by power uprate. In core design increased average bundle enrichment and batch fraction may be employed to increase cycle energy.