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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CONCERNING GENERAL ELECTRIC LICENSING TOPICAL REPORT

NEDC-31984P

GENERIC EVALUATIONS OF GENERAL ELECTRIC

BOILING WATER REACTOR POWER UPRATE, VOLUMES I AND II

1.0 INTRODUCTION

By letter dated July 30, 1991, General Electric (GE) Nuclear Energy submitted Licensing Topical Report (LTR) NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Volumes I and II. Volume I of the LTR contains generic bounding analyses and equipment evaluations in support of power uprate amendment requests, as well as an overview of the impact of the uprated conditions on licensees' responses to various NRC and industry generic correspondence. Volume II of the LTR contains a detailed listing of the generic correspondence reviewed, as well as GE's determination of the potential impact of uprated conditions on the topics discussed in that correspondence. A non-proprietary version of this LTR was submitted in March of 1992, at the staff's request. This evaluation addresses only the proprietary version of the LTR, which was submitted in July of 1991; however, the topic discussions provided in the non-proprietary version, although not as detailed, provide similar information to that evaluated by the staff.

The program for the Boiling Water Reactor (BWR) power uprate effort is contained in GE LTRs titled "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," NEDC-31897P (Proprietary) and NEDO-31897 (Non-proprietary). The NRC staff has previously approved the proposed program, as described in NEDC-31897P, in a Staff Position dated September 30, 1991. These documents include guidance to individual licensees as to the scope and content of information to be submitted as a part of plant specific power uprate submittals.

2.0 EVALUATION

The NRC staff has reviewed GE LTR NEDC-31984P and has found that the LTR, when combined with sufficient additional plant-specific information, constitutes an adequate basis for the review of individual licensee power uprate amendment requests. This safety evaluation documents the staff's review of the generic analyses and evaluations provided in the LTR. In certain instances, the staff has determined that additional information will be required from plant specific submittals in order to complete our evaluation of the subjects discussed in the LTR. The additional requirements for these subject areas are clearly delineated in the applicable sections of this evaluation.

A detailed evaluation of each subject area discussed in the LTR follows.

2.1 GE Review of Generic Communications

An audit of GE's assessment of the impact of uprate on licensee responses to generic NRC and industry communications was performed in December, 1991. This audit reviewed GE's procedures for assessing the impact of uprate on the topics discussed in each piece of generic correspondence, as well as evaluating selected items from the list of generic correspondence included in Volume II of the LTR. The staff concluded that the list of generic communications provided in Table 2-1 of Volume I of the LTR is a complete and thorough listing of those items potentially affected by a power uprate. Furthermore, GE has provided generic bounding analyses and evaluations for all items which are designated as "BWR Bounded" in Table 2-1. Therefore, licensees applying for a power uprate will only be expected to address the generic correspondence listed in the table as "Plant Dependent", as well as generic correspondence published since December, 1991.

2.2 GE Setpoint Methodology

The staff will issue a Safety Evaluation concerning the GE Instrument Setpoint Methodology, NEDC-31336, as part of a separate review effort (TAC No. M81253). The staff has found that the methods described in that LTR for calculating trip setpoints for a variety of instrument types are acceptable. Therefore, use of the methods described in NEDC-31336 for recalculating setpoints which are affected by parameter changes associated with power uprate is acceptable. The staff will audit sample data sheets and calculations for plant specific setpoint changes. Each licensee will be contacted after submitting their uprate submittal and instructed to submit specific data sheets and calculations.

2.3 Emergency Operating Procedures

As a result of power uprate, a number of variables and limits utilized in plant specific Emergency Operating Procedures (EOPs) may be affected. In particular, the increase in rated reactor power will directly or indirectly affect many of the variables and limit curves contained in the existing plant EOPs. Although the conditions which require operator actions contained in the EOPs may change, the operator actions described in the EOPs will remain the same after uprate.

The LTR includes a table which describes the particular EOP value recalculations which will be required as a result of power uprate. The table is based upon Revision 4 of the Emergency Procedure Guidelines (EPGs) which have been developed by the Boiling Water Reactor Owners' Group (BWROG). Each licensee will be required to recalculate their plant specific EOP variables and limit curves using the guidance provided in the LTR, as well as to ensure that the EOPs are modified to address changes to other process variables which are not addressed in the LTR. However, licensees will not be required to submit the revised EOPs as a part of their power uprate submittals.

2.4 Generic Analytical Evaluations

As a part of the generic power uprate effort, GE committed to perform generic bounding analyses and equipment evaluations for selected pieces of vendor (GE) supplied Nuclear Steam Supply System (NSSS) equipment which is similar between all, or groups of, BWRs. Staff review of those analyses presented in the LTR is documented below.

2.4.1 Loss of Feedwater Flow Transient

The results of loss of feedwater flow (LOFW) transient analyses for all classes of operating reactors involved in the power uprate program (BWR/4 through BWR/6) were presented. These analyses demonstrated that, for all plants, the original design bases of the reactor core isolation cooling (RCIC) system for maintaining water level above the top of the active fuel (TAF) were preserved during a loss of normal feedwater when the other, higher capacity high pressure water supply system was assumed to have failed. The bounding analyses for BWR/4, BWR/5, and BWR/6 product line plants were presented by product line. The minimum water levels for this transient with uprated conditions were compared to the minimum water levels for the original licensing basis of these same plants. In the limiting case, a 218 inch diameter BWR/4 vessel, the analysis demonstrated that at least five feet of water would remain above the TAF.

The worst case LOFW transient analysis shows a reduction of the minimum water level from approximately ten feet above TAF using the original licensing basis power level and calculational methodology, to a level not less than five feet above the TAF when using the uprated power level and additional conservative assumptions (including delayed RCIC initiation). The water level outside the shroud remains above the ECCS initiation setpoint for the uprated case. The results of these analyses are acceptable in that the original licensing basis is met; i.e., water level inside the shroud is maintained above TAF. However, the reduced water levels may have an impact on the amount of time available for operator intervention during a LOFW coincident with additional equipment failures. Therefore, plant specific submittals should address the impact of the reduced water level on operator action times for the LOFW transient with additional failures.

2.4.2 Core Thermohydraulic Stability

The BWROG and the staff are currently addressing methods to minimize the occurrence and potential effects of core power oscillations which have been occasionally observed during certain BWR operating conditions; particularly, during plant operations with low core flow on a high power rod line. Until this issue is resolved, individual plant specific submittals must adopt the operational constraints described in the LTR, which are consistent with guidance provided in NRC Bulletin 88-07 and Supplement 1 to that Bulletin. Specifically, these operational constraints will continue to restrict plant operation in the high power rod line, low core flow region of the power/flow map. The operator actions required upon entry into the regions of the power/flow map identified as having a low stability margin shall remain as specified in Supplement 1 to Bulletin 88-07.

2.4.3 Core Spray Distribution

The applicability of the core spray distribution assumptions for power uprate conditions utilized in the GE Loss-of-Coolant Accident/Emergency Core Cooling System (LOCA/ECCS) models was addressed in the report. In the short term following a postulated LOCA, no credit is given for core spray flow to high power fuel bundles until the upper plenum region forms a pool of water covering the upper tie plate of all fuel bundles. The drainage flow rate to the high and average power fuel bundles is determined by counter current flow limiting (CCFL) characteristics, further reducing the credit given for core spray. The model allows for CCFL breakdown in the peripheral region of the core after the upper plenum water level rises above the core spray sparger. When the CCFL breakdown occurs in the peripheral bundles, a rapid drainage of water occurs from the upper plenum to the lower plenum through the peripheral bundles, supporting a reflooding of the core. This evaluation methodology results in very little credit being given for spray cooling during the short term response to a postulated LOCA.

Since the increase in core power for uprate is accomplished by flattening the radial bundle power profile, assurance must be given that the assumption of CCFL breakdown in peripheral bundles will remain valid. Individual licensees should adhere to existing radial power shape limitations when designing core reloads for uprated conditions. Provided that the radial power distribution remains within the bounds of the LOCA/ECCS assumptions, the effect of power uprate on the short term response to a postulated LOCA should be minimal.

In the longer term, credit for spray cooling is given while water level in the core shroud remains below TAF. For these conditions, at least one core spray loop is assumed to be operating. Test data for the verification of core spray distribution is based on the short term portion of the accident analysis when power levels and steam generation from the core (or from reactor depressurization) are much higher than in the long term portion of the analysis. Therefore, steam generation during the long term portion of the accident will be much less severe than during the short term portion, and the effect of core spray distribution on the long term response to a postulated LOCA is bounded by the effect on the short term response.

The effects of power uprate on short term response are addressed in the GE LOCA/ECCS models and plant specific submittals will utilize these models to show compliance with the criteria described in 10 CFR 50.46. The impact of power uprate on the long term response to a LOCA will continue to be bounded by the short term response.

2.4.4 Safety Limit Minimum Critical Power Ratio

Plant specific reload submittals shall contain analyses to confirm that the safety limit minimum critical power ratio (SLMCPR) is appropriate for the uprated average bundle power. This will be done by comparing bundle power to the applicable SLMCPR basis in NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel (GESTAR)," U.S. Supplement. If a new plant specific SLMCPR is needed because the uprated core average bundle power exceeds the documented licensing basis, it will be established using the same NRC approved procedures and will be included in the plant specific submittal.

2.4.5 Containment Atmosphere Combustibility

Title 10 of the Code of Federal Regulations, Section 50.44 (10 CFR 50.44) required licensees to install means to control hydrogen gas that may be generated following a postulated LOCA. The LTR assumed that fuel assemblies utilized for power uprate would have no significant difference in the amount of cladding material, and hence the total amount of hydrogen gas generation during a postulated LOCA. Based on this assumption, the LTR concluded that power uprate would not affect the design basis metal-water hydrogen generation source term used by individual plants in the design of installed combustible gas control equipment.

The post-LOCA containment atmosphere combustibility is also affected by changes in the concentration of free oxygen produced by radiolytic decomposition of water in the reactor. An increase in reactor power will result in a slight increase in the amount of oxygen liberated by radiolytic decomposition of water, which is directly proportional to reactor power level. The LTR concluded that this small increase in oxygen concentration is well within the capacity of currently installed combustible gas control systems. Plant specific submittals will confirm the capability of the combustible gas control system, and will address any procedural or equipment setpoint changes which may be required to assure adequate containment atmosphere combustible gas control.

2.4.6 Materials and Coolant Chemistry

Both the NRC staff and the industry have expressed concern regarding the occurrence of intergranular stress corrosion cracking (IGSCC) in austenitic stainless steel piping and components. In response to these concerns, programs for the periodic monitoring, inspection, and replacement of affected components, as well as programs governing materials selection and water chemistry have been developed by the industry. The potential effect of uprated conditions (increased temperatures, pressures, and flow rates) will not significantly impact the occurrence of IGSCC. Therefore, programs currently in place for the mitigation of IGSCC should not be significantly affected. Licensees will be required to continue to meet commitments made in response to NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping" and its Supplements.

2.5 Generic Hardware Capability Evaluations

As an additional incentive for NRC and industry participation in the generic BWR power program, GE committed to perform generic evaluations of significant NSSS equipment. Staff comments regarding these evaluations are shown below.

2.5.1 Low Pressure Emergency Core Cooling Systems

In the LTR, GE provided a generic bounding evaluation of the performance and design requirements of the low pressure emergency core cooling systems (ECCS).

This evaluation assumed initial reactor operating conditions commensurate with power uprate; namely, a 4.3 percent increase in reactor power, a 40 psig increase in reactor vessel dome pressure, a 5°F increase in reactor saturation temperature, and a 5 percent increase in steam and feedwater flow rates.

The evaluation concluded that the operational conditions for the low pressure ECCS will not be affected by power uprate. The pressure setpoints for the low pressure coolant injection mode of the residual heat removal (RHR/LPCI) and low pressure core spray (LPCS) systems will not be changed for power uprate, so these systems will not experience increased operating pressures. The licensing and design flow rates for the low pressure ECCS will not be increased. In addition, the RHR system shutdown cooling mode flow rates and operating pressure will not be increased. Since these systems do not experience different operating conditions, there is no significant impact on the operation of these systems from power uprate, except for a possibly longer cooldown time.

2.5.2 <u>High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System</u>

The design bases for the HPCI and RCIC systems are to provide reactor vessel inventory control during (1) small and intermediate break size LOCAs (HPCI with other ECCS as backup), and (2) transients involving loss of feedwater flow (RCIC with HPCI as backup). These systems are designed to provide their rated flows over a reactor vessel pressure range from 150 psig to a maximum pressure based on the lowest opening setpoint for the safety relief valves (SRVs). The opening setpoints for the SRVs will be increased to maintain an adequate simmer margin above the increased reactor operating pressure associated with power uprate. Increasing the SRV pressure setpoints will have a potential impact on the maximum operating pressure for the HPCI and RCIC systems for reactor isolation events. The assumed increase in reactor operating pressure of 40 psig results in an approximate 3 percent increase in required pump discharge pressure.

The required HPCI and RCIC water flow rates will remain unchanged after uprate. However, the pump and turbine operational requirements are increased due to the increased SRV pressure setpoints (as described above). The required increase in HPCI and RCIC discharge pressures will require the turbine speeds to be increased slightly. This change in turbine speed will not significantly affect the operation of the HPCI and RCIC systems. Both the HPCI and RCIC systems are capable of meeting the flow requirements at the increased pressures associated with uprate.

The LTR assessed the impact of increased reactor pressure on the potential for turbine overspeed during startup of the HPCI and RCIC systems. The increased reactor operating pressure associated with uprated conditions has the potential to result in increased turbine overspeeding during system startup, increasing the probability of the system to trip. The LTR stated that modifications to HPCI systems which use Terry Corporation turbine assemblies will be made as described in GE Services Information Letter (SIL) No. 480. Likewise, modifications to the RCIC system will be made as described in GE SIL No. 377.

In order to reduce the possibility of turbine overspeed trips, plant specific submittals must address the modifications described in GE SIL No. 480 and GE SIL No. 377 (or equivalent modifications). Plant specific submittals must also provide assurance that the HPCI and RCIC systems will be capable of injecting their design flow rates at the higher reactor operating pressures associated with power uprate. Additionally, each licensee must also provide assurance that the reliability of these systems will not be decreased by the higher loads placed on the systems or because of any modifications made to these systems to compensate for these increased loads.

2.5.3 High Pressure Core Spray (HPCS) System

The HPCS system on BWR/5 and BWR/6 plants consists of a single, motor-driven centrifugal pump located outside of the primary containment, a peripheral ring spray sparger located in the reactor vessel above the core region, and associated piping, valves, controls and instrumentation. The system is designed to operate from normal offsite auxiliary power or from an emergency diesel generator if offsite power is not available. The primary purpose of the HPCS system is to maintain reactor vessel water level inventory during a postulated small break LOCA that does not immediately depressurize the vessel. The HPCS also serves as a backup to the RCIC system for the loss of feedwater transient for BWR/5 and BWR/6 product line plants. The HPCS system was designed to provide makeup water over the entire range of reactor operating pressures, and the physical equipment design of the HPCS system is compatible with a reactor maximum design pressure of 1250 psig, which bounds the potential range of HPCS system operating pressures.

Increased reactor vessel operating pressures have little impact on HPCS system effectiveness during a postulated large break LOCA since the primary contribution of HPCS to core cooling would occur during and following reactor depressurization. For postulated small and intermediate break LOCAs, peak fuel cladding temperatures will increase slightly due to lower HPCS flow rates caused by increased reactor vessel pressures. The effect of uprate on HPCS flow and resultant peak cladding temperatures for small and intermediate break LOCAs will be verified on a plant and fuel bundle specific basis and will be documented in the plant specific uprate submittal.

The HPCS flow rate is approximately three times larger than the RCIC flow rate for all BWR/5 and BWR/6 plants. Therefore, the plant response to a loss of feedwater flow transient with only HPCS available is bounded by the plant response to the LOFW with only RCIC available. (See Section 2.4.1.)

2.5.4 Control Rod Drives and Scram Performance

The increased reactor vessel dome pressure associated with power uprate produces a corresponding increase in the bottom head pressure. This increase in reactor pressure has a small effect on the insertion times for control rods during a scram. For pre-BWR/6 product line reactors, initial control rod scram insertion speeds are slowed due to the increased reactor pressure. However, near the end of the control rod stroke, the increased reactor pressure will speed up the control rod insertion rate, resulting in a slightly shorter overall scram time for the control rods. Pre-BWR/6 plants will be required to provide assurance that the scram time performance indicated in the current plant Technical Specifications will be maintained.

Due to a difference in control rod drive (CRD) system design, BWR/6 plants may experience slightly longer control rod scram stroke times after uprate. This anticipated change in scram performance may require changes to the plant Technical Specification requirements for scram stroke times. Plant specific submittals 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 Technical Specifications.

Normal control rod insertion and withdrawal functions will not be significantly affected by the increase in reactor bottom head pressure. Normal control rod drive header pressure is maintained approximately 250 psig above the lower head pressure. Plant specific submittals will address the ability of the CRD system to maintain an adequate pressure differential for control rod operation.

2.5.5 Recirculation System

Reactor pressure will be increased approximately 40 psig, which causes a 5°F increase in the reactor vessel saturation temperature. These increases are small when compared to the original design operating conditions of approximately 1000 psig and 540°F. Licensees will be expected to review plant specific operating data to assure that the recirculation system will accommodate the small increase in flow resistance which is expected due to the increase in core average void fraction due to uprate. The results of this review will be documented in the plant specific uprate submittal. An evaluation of recirculation system vibration will also be included in the plant specific submittal. The licensee must ensure that the recirculation system, as well as other pressure boundary components or systems, continue to meet ASME Code requirements.

2.5.6 Safety Relief Valves (SRVs)

The LTR stated that the performance of BWR safety relief valves was evaluated using the conditions associated with power uprate, such as higher reactor steam flow, higher operating pressure, and higher operating temperature. The LTR concluded that the increased steam flow should not affect the SRVs, which are normally closed during plant operations. The opening transient for the SRVs under higher steam flow conditions will not be significantly different from the present opening transients. The existing SRVs will have sufficient capacity to accommodate transients which occur from the uprated power level. Plant specific submittals will be required to confirm the capability of the SRVs to meet ASME Code requirements for reactor vessel overpressure protection.

To provide sufficient simmer margin, the SRV valve spring opening setpoint pressures will be increased proportionally to the increased operating pressure for uprated conditions. Procedures currently used for the recertification of SRVs will require revision to provide testing under the higher normal operating pressure. Pressure switches, used in some plants to open SRVs during pressure transients, will be reset to accommodate the higher operating pressure. The pressure switch setpoints will be chosen high enough to limit SRV actuations during minor transients, yet low enough to provide the relief action assumed in transient analyses. Readjustment of SRV valve springs and/or pressure switches will be addressed in plant specific submittals.

Integrity of the SRVs will not be affected by uprated conditions. All safety relief valves currently in use have design pressures of at least 1250 psig. The small increase in reactor pressure associated with power uprate (approximately 40 psig or less) will not impact the structural integrity of the SRVs. Additionally, the small increase in reactor temperature (approximately 5°F) will not affect the operation of the SRVs.

Increasing SRV setpoints will involve increases in the peak reactor coolant system pressures and temperatures. These increased parameters will result in higher differential pressures and flow rates for the operation of safety related valves in systems connected directly to the reactor coolant system boundary. These systems include the HPCI, HPCS, RCIC and RWCU systems. Licensees will need to demonstrate that the safety related valves in these systems are capable of operating against the worst case design-basis conditions in light of the more severe pressures and temperatures. Because of the inability to test these valves under differential pressure and flow conditions in the plant, sufficient margin must be provided in evaluations of the capability of these valves.

2.5.7 Main Steam Isolation Valves (MSIVs)

Main steam isolation valves, as well as being a part of the reactor coolant pressure boundary, provide a number of safety functions, including the isolation of the reactor from the environment during postulated accident scenarios. Because of this, the current licensing basis requirements for the MSIVs must continue to be met under uprated conditions. The impact of increased pressures and temperatures on the MSIV structural integrity is minimal, because such changes are very small in comparison to the original design specifications of the valves. The increase in fatigue duty due to increased pressure and temperature is insignificant due to the very low fatigue usage of the MSIVs. Additionally, the reactor coolant pressure boundary requirements of the MSIVs, such as closure time and leakage limits will continue to be monitored by various surveillance requirements in the plant Technical Specifications to ensure that the original licensing basis for the MSIVs is preserved.

The increased normal operating steam flow associated with power uprate will cause the MSIVs to close slightly faster after uprate, due to increased assistance in closure provided by the additional steam flow. This increased closure speed could increase the impact loading of the valves when closing from rated flow conditions. However, the MSIVs are designed to withstand the closure impact from 200% of original rated steam flow. Therefore, the increased impact loading of the MSIVs due to uprate are small, and are bounded by the original design.

Faster closure times for the MSIVs during transient and accident conditions have the potential to impact peak reactor vessel pressures and power levels. Licensees must ensure compliance with existing Technical Specification limits for MSIV closure times (both upper and lower bound) after uprate.

The class 1E components, such as MSIV limit switches and solenoid valves, could potentially be affected by the higher reactor operating temperature associated with uprate, due to their proximity to the MSIV valve bodies. It is necessary that the environmental qualification (EQ) design conditions for these components bound potential operating conditions after uprate. Licensees will be expected to confirm the qualification of these components on a plant specific basis in order to assure that potential accident conditions are bounded.

2.6 Impact on Safety Margins

Also included in the LTR was a generic assessment of the impact of power uprate on plant safety margins. Although this generic discussion is not sufficiently detailed to exclude individual licensees from the need to assess the impact of uprate on their facilities, it does provide assurance that the expected changes to current safety margins will be small. The underlying philosophy of the generic BWR power uprate program has been that any changes to plant safety margins which result from uprate must be acceptably small. The discussion contained in the LTR confirms GE's commitment to this philosophy. Plant specific uprate submittals will also address the impact of power uprate on plant safety margins.

2.6.1 Fuel Thermal Limits

No change is required to the basic fuel design to achieve the uprated power level or to maintain the margins as discussed in the LTR. No increase in the allowable peak bundle power has been requested. A slightly flatter radial power distribution may be utilized to supply the additional power and maintain limiting fuel bundles within current constraints. The fuel operating limits, such as maximum average planar linear heat generation rate (MAPLHGR) and operating limit minimum critical power ratio (OLMCPR), will still be met at the uprated power level. The plant-specific submittal will confirm the acceptability of these operating limits as determined for uprated power conditions. Reload analyses will continue to meet acceptable NRC criteria as specified in GESTAR. New fuel designs will need to meet NRC approved acceptance criteria. GE fuel will continue to meet the criteria accepted by the NRC as specified in NEDO-31908, "Licensing Criteria for Fuel Designs."

2.6.2 Design Basis Accidents

Plant specific analyses will continue to demonstrate the ability of each plant to cope with the full spectrum of hypothetical pipe break sizes in the largest recirculation, steam, feedwater and ECCS lines, through breaks as small as instrument lines. These analyses will deal with both high and low energy line breaks, as well as the success of plant systems in dealing with the breaks while accommodating a single active equipment failure in addition to the break. Challenges to the fuel and containment, as well as potential radiological releases to the environment, will be assessed on a plant specific basis using NRC approved methods.

2.6.3 Transient Evaluations

The effects of plant transients will be analyzed against the safety limit minimum critical power ratio (SLMCPR) which will be established using NRC approved procedures, as described in Section 2.4.4 of this evaluation. The SLMCPR will be confirmed for each plant requesting a power uprate. Transient events will continue to be analyzed against this SLMCPR, using NRC approved procedures, when establishing the operating limit minimum critical power ratio (MCPR). This operating limit MCPR will be documented in each plant specific uprate submittal and confirmed for each cycle of operation in the cycle-specific reload analysis.

3.0 ENVIRONMENTAL CONSEQUENCES

The environmental effects of power uprate, as discussed in the LTR, will be controlled at the same limits as are currently used at each plant. The present limits for plant environmental releases, such as ultimate heat sink temperature or plant vent radiological limits, will not be increased for operation at uprated conditions. Therefore, no significant environmental impact is expected to occur as a result of power uprate. Plant specific submittals will address any deviations from the discussion provided in the LTR.

4.0 CONCLUSION

The staff has reviewed the information presented in GE Licensing Topical Report NEDC-31984P, and has determined that the generic bounding analyses and equipment evaluations contained in the LTR do provide sufficient information to evaluate certain aspects of individual plant response to power uprate. Some of the topic areas discussed in the LTR will require either additional plant specific information or verification before the staff will be able to complete its evaluation of those topics. Requirements for additional information needed to complete staff review are documented throughout this evaluation. Those topic areas will be resolved in individual plant specific safety evaluations. The staff considers its review of GE LTR NEDC-31984P, Volumes 1 and 2, to be completed. Staff review of supplements to the LTR will be documented in supplemental staff Safety Evaluations.

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