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MEMORANDUM FOR: L. B. Marsh, Project Director (13D-18)
Project Directorate III-1
Division of Reactor Projects III, IV, V

FROM: Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Technology

SUBJECT: GENERIC EVALUATIONS OF GENERAL ELECTRIC BOILING WATER REACTOR POWER UPRATE, VOLUME 1 NEDC-31984P Topical Report Review (TAC NO. M81253)

Enclosed is the Reactor Systems Branch input to the Safety Evalaution Report being prepared by your Project Directorate for the subject topical report. It is my understanding that you will use this and other technical branch inputs for developing the overall staff safety evaluation for this generic topical report.

original signed by
Robert C. Jones

Robert C. Jones, Chief
Reactor Systems Branch
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Enclosures:
As stated

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SAFETY EVALUATION REPORT

3.0 Analytical Evaluations

3.1 Loss of Feedwater Flow Transient

Generic Electric provided analytical evaluations of the generic aspects of power uprate in report NEDC-31984P, Volume 1 and Volume 2. In particular, the Loss of Feedwater Flow (LOFW) transient results for all classes of operating reactors was presented. These evaluations were presented to show that the original design basis for the reactor core isolation cooling (RCIC) for maintaining water level above the top of the active fuel (TAF) were preserved during a loss of normal feedwater with the other higher capacity high pressure water supply system assumed failed. The bounding analyses for BWR/4, BWR/5 and BWR/6 plants is presented. These reductions in water level with uprated power are compared to the maximum water level reduction for the original licensing basis of these same plants. In the limiting case, a 218 inch BWR/4 vessel, at least five feet of water remain above the TAF. The worst case LOFW transient shows a reduction of the minimum water level from approximately ten feet above TAF (original licensing basis) to greater than five feet above TAF for the uprate power case with conservatively assumed delayed mitigative features such as RCIC initiation. The water level outside the shroud remains above the level one ECCS initiating setpoint. Even though the results of the decreased water level are acceptable in meeting the original licensing basis of the plant, i.e. water level above TAF, the plant specific evaluation that references the generic results should provide a discussion on the reduction of safety margin associated with the reduced water level resulting from power uprate when compared to the original licensing basis. The transient results with relaxed parameters should be presented separately. The plant specific submittals should also address the impact of a reduced water level on operator action times for the LOFW transient with additional failures.

3.2 Stability

The BWR Owner's Group and the NRC are addressing ways to minimize the occurrence and potential effects of oscillations that have occasionally been observed for certain BWR operating conditions. Until a resolution is developed, procedures have been incorporated in accordance with NRC Bulletin 88-07 and Supplement 1 to that bulletin which restrict plant operation in the high power, low core flow region of the power/flow operating map for power uprate. Plant specific submittals will adopt these procedures to develop their power/flow operating map.

3.3 Core Spray Distribution

The applicability of the core spray distribution analysis assumptions at power uprate conditions, used in the GE LOCA/CCCS models was addressed in the NEDC-31984P generic report. In the short term, 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 power bundles and average core is determined by counter current flow limiting (CCFL) characteristics further reducing credit for core spray. The model allows CCFL breakdown in the peripheral region of the core after the upper plenum water level raises above the core spray sparger. When the peripheral CCFL breakdown occurs, there is a rapid drainage of water from the upper plenum to the lower plenum through the peripheral bundles of the core, supporting reflooding of the core from the lower plenum. This results in very little credit for core spray cooling during the short term response to a postulated LOCA. Since a power uprate results in a radial bundle power profile that is flatter across the core, plant specific submittals must provide assurance that the codes used to predict CCFL breakdown through peripheral lower power bundles during a postulated LOCA are valid for the bundle powers and distributions associated with power uprate.

In the longer term, credit for core spray is taken while the water level is below the TAF. For these conditions, at least one core spray loop will be operating. Test data for verifying core spray distribution is based on the short term portion of the transient when power levels and steam generation from the core, or depressurization, are much higher than in the long term portion of the transient. Therefore, any steam generated in the long term is less severe than the steam generated in the short term tests which are used to verify core spray distribution, even at uprated power.

The short term effects of power uprate are addressed in the GE LOCA/ECCS models and plant specific submittals will utilize these models to show compliance with the 10 CFR 50.46 criteria, because the models do not take credit for core spray in the short term at the higher steam generation rates. The impacts of power uprate on core spray distribution during long term cooling are bounded by the short term tests at the higher steaming rates.

3.4 Safety Limit Minimum Critical Power Ratio

Plant specific 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 GESTAR (NEDE-24011-P-A-10-US," General Electric Standard Application for Reactor Fuel (GESTAR)," U.S. Supplement, March 1991). 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 be included in the plant specific submittal.

4.1 Low Pressure Emergency Core Cooling System (ECCS)

The operational conditions for the low pressure ECCS will not be affected by power uprate. The pressure set points of the residual heat removal (RHR), and the low pressure coolant injection (LPCI) modes of operation will not be changed for power uprate therefore these systems will not experience higher operating pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. In addition, the RHR shutdown cooling mode flow rates and operating pressure will not be increased. Since these systems do not experience different operating conditions, there is no impact due to power uprate, except for a possible longer cooldown time.

4.2 High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems

The HPCI and RCIC system design bases are to provide reactor vessel inventory makeup during small and intermediate break loss of coolant accidents (HPCI only with other ECCS as backup), and transients involving loss of feedwater flow (RCIC with HPCI as backup).

The HPCI and RCIC systems are designed to provide their rated flows over a reactor vessel pressure range of 150 psig to the maximum pressure based on the lowest safety relief valve (SRV) spring safety setpoint. The SRV opening setpoints will be increased for power uprate to maintain adequate simmer margin. Increasing the SRV setpoint pressure has a potential impact on the maximum operating pressure for the HPCI and RCIC systems (for isolation events).

4.2.1 Assessment of HPCI Performance

The required HPCI water flow rate remains unchanged. However, the HPCI pump and turbine operational requirements at uprated conditions are increased; the pump total dynamic head by approximately three percent due to an SRV setpoint increase, and new speed and power requirements by the steam turbine (increased steam flow rate and inlet control valve steam pressure).

4.2.2 Assessment of RCIC Performance

The RCIC operational requirements were reviewed in the same manner as described above for the HPCI system. The RCIC system design flow rate remains unchanged and similar requirements are required of the pump and the turbine, as in the HPCI system.

4.2.3 Assessment of Turbine Overspeeding

The assessment of turbine overspeeding is described in the topical report because the startup transient for the HPCI and RCIC systems are at a potentially higher inlet pressure which may result in increased turbine overspeeding, increasing the probability of the system to trip. Modifications will be made to the HPCI systems with Terry Corporation turbine assemblies as described in GE Services Information Letter (SIL) No. 480. Modifications to the RCIC system will be made as described in GE SIL No. 377. In order to avoid the possibility of turbine overspeed trips, plant specific submittals must address the modifications addressed in GE SIL No. 480 and GE SIL No. 377 (or equivalent modification) as part of their power uprate program.

4.2.4 Conclusion

Licensee plant specific submittals for power uprate must provide assurance that their HPCI and RCIC systems are capable of injecting their design flow rates at the higher operating parameters associated with power uprate. Licensees must also provide assurance that the reliability of these systems will not be decreased because of the higher loads placed on the systems or because of any modifications made to these systems to compensate for these increased loads.

4.3 High Pressure Core Spray (HPCS) System

The HPCS systems on BWR/5 and BWR/6 plants consists of a single, motor-driven centrifugal pumps (located outside the primary containment), a peripheral ring spray sparger in the reactor vessel located above the core, and associated piping, valves, controls and instrumentation. The system is designed to operate from normal offsite auxiliary power or from a standby diesel generator (if offsite power is not available).

The primary purpose of the HPCS is to maintain reactor vessel water inventory for the 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 flow transient.

The HPCS systems were designed to pr de makeup water over the entire operating pressure range and the p ysical equipment is designed compatible with the reactor vessel design pressure of 1250 psig, which bounds its potential range of operation.

4.3.2 Post LOCA Performance

The increased vessel operating pressure has little influence on HPCS effectiveness for a large LOCA since its primary role is after depressurization has occurred. The effect on HPCS flow and the resultant peak cladding temperature, due to increased vessel operating pressure for the small break LOCA will be verified on a plant/fuel bundle specific basis and will be documented in the licensee plant specific submittal.

For the loss of feedwater transient with one system failed, GE calculations show that the lost inventory will be greater for the RCIC alone case than for the HPCS alone case.

4.4 Control Rod Drives and Scram Performance

The increased dome pressure due to power uprate produces a corresponding increase in the bottom head pressure. For pre-BWR/6 plants, the initial control rod insertion is slowed down due to the increased pressure. However, near the end of the scram stroke, the higher reactor pressure will actually speed up control rod insertion. For plants with BWR/6 control rod drives (CRDs), nominal scram times may be slightly longer, since the accumulator pressure is working to overcome the higher vessel pressure due to power uprate. Plant specific submittals must provide assurance that the scram insertion speed used in the BWR/6 transient analysis is slower than the scram insertion speed contained in the Technical Specifications and verified by periodic surveillance measurements. Pre-BWR/6 plants have a slightly shorter scram time due to the increased pressure, however, they also must provide assurance of the scram time performance indicated in the plant specific Technical Specifications.

4.5 Recirculation System

An increase of approximately 4.3% in power will be accompanied by an increase of 40 psi and 5°F at the reactor coolant pressure boundary. These increases are small when compared to the original operating conditions of 1000 psig and 540°F. A review of plant-specific operating data to assure that the recirculation system will accommodate the expected small increase in flow resistance due to the increase in core average void fraction at the uprated condition when operating at maximum core flow will be performed. The results will be documented in the plant-specific licensing report. Evaluation of recirculation system vibration due to the potential for increased flow will also be addressed in the plant specific report. The recirculation system, as well as other pressure boundary system or components, must continue to meet the ASME code requirements.

4.6 Safety Relief Valves

The performance of BWR safety relief valves (SRV) was evaluated under the conditions of power uprate such as higher steam flow (5%), higher operating pressure (+40 psi), and higher temperature (5°F). The increase steam flow should not affect the SRVs, since the valves are normally closed and the opening transient is not significantly different for transients initiated from higher steam flow conditions. The existing SRVs must have sufficient capacity to accommodate transients which occur from uprated power. Specific plant submittals will be required to confirm this capability to meet ASME code requirements for overpressure.

To ensure adequate simmer margin, the valve spring opening setpoint pressure will be increased proportionally to the operating pressure. Procedures currently used for recertification of SRVs will require revision to provide testing at the higher normal operating pressure. Pressure switches, which are used in some plants to open SRVs during pressure transients, will require resetting. The pressure switch setpoints will be chosen high to limit SRV actuations under minor transients, yet low enough to provide the relief action taken credit for in transient analyses. The above mentioned items will be addressed by licensees in their plant specific submittals for power uprate.

4.7 Main Steam Isolation Valves (MSIVs)

The reactor coolant pressure boundary requirements of the MSIVs such as closure time and leakage will continue to be monitored by various surveillance requirements in the plant Technical Specifications to ensure the original licensing basis for the MSIVs is preserved.

The class IE components such as MSIV limit switches and solenoid valves, could be potentially affected by the slightly higher operating temperature, due to power uprate. It is necessary that the design conditions for these components bound the power uprate conditions. This must be confirmed on a plant specific basis to assure potential accident conditions are bounded.

5.0 Impact on Safety Margin

5.1 Fuel Thermal Limits

No change is required in the basic fuel design to achieve uprated power level or to maintain the margins as discussed in this report. No increase in allowable peak bundle power is requested. A slightly flatter radial power distribution may be utilized to supply the additional power and still maintain limiting fuel bundles within their present constraints. The fuel operating limits such as maximum average planer linear heat generation rate (MAPLHGR) and operating limit minimum critical power rate (OLMCPR) will still be met at the uprated power level. The plant-specific submittal will confirm the acceptability of these operating limits as set for uprated power operation. Reload analyses will continue to meet the acceptable NRC criteria as specified in NEDE-24011-P-A-10-US, "General Electric Standard application for Reactor Fuel (GESTAR)," US Supplement, March 1991. New fuel designs will meet NRC approved acceptance criteria. GE fuel will meet the criteria accepted by the NRC as specified in NEDO-31908, "Licensing Criteria for Fuel Designs," January 1991.

5.2 Design Basis Accidents

The BWR licensing evaluations will be continued by demonstrating the ability for coping with the full spectrum of hypothetical pipe break sizes in the largest recirculation, steam, feedwater, and ECCS lines, down to breaks as small as instrument lines. This break spectrum analytically investigates a full spectrum of large and small, high and low energy line breaks, and the success of plant systems in dealing with them while accommodating a single active equipment failure in addition to a postulated LOCA. Challenges to the fuel, containment, and the radiological releases will be assessed on a plant specific basis.

The plant specific analyses will include challenges to the fuel by calculating the fuel peak cladding temperature (PCT) along with evaluating the other 10 CFR 50.46 acceptance criteria. The challenges to the containment that are impacted by power uprate include plant specific containment pressure and temperature. Containment dynamic loads which may be affected by power uprate will also be evaluated in the plant specific submittal.

5.3 Transient Evaluations

The effects of plant transients are evaluated against the safety limit minimum critical power ratio (SLMCPR) which is a limit that is established using NRC approved procedures discussed elsewhere in this evaluation. The SLMCPR will be confirmed for each plant requesting a power uprate. Transient events will continue to be evaluated against this SLMCPR, using NRC approved procedures, for establishing the operating limit MCPR. This operating limit MCPR will be documented in each plant specific uprate submittal and confirmed for each cycle of operation in the reload analysis.