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SUBJECT: Provides addl info requested by NRC in support of TS-384. Encl 1 provides util response to 980313 NRC RAI for 971001, proposed TS change.

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Christopher M. (Chris) Crane
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May 20, 1998

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

Gentlemen:

In the Matter of	)	Docket Nos.	50-260
Tennessee Valley Authority	)		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING UNITS 2 AND 3 TECHNICAL SPECIFICATION (TS) CHANGE TS - 384, REQUEST FOR LICENSE AMENDMENT FOR POWER UPRATE OPERATION (TAC NOS. M99711 AND M99712)

This letter provides additional information requested by NRC in support of TS-384. On October 1, 1997, TVA provided TS-384, an amendment to Operating Licenses DPR-52 and DPR-68 that will allow Units 2 and 3 to operate at an uprated power level of 3458 MWt.

Enclosure 1 provides TVA's response to the March 13, 1998, NRC RAI for the October 1, 1997, proposed TS change. This letter includes replies to NRC's requests except for Section C, "Environmental Qualification for Safety-Related Electrical Equipment."

The post-accident environmental evaluations for uprated conditions which are necessary to reply to Section C have not been completed. Preliminary evaluations performed for equipment located inside the drywell indicate that the drywell accident profile for uprated conditions are enveloped by the equipment test profiles. Following the completion of

D030

these evaluations for affected areas in the plant, TVA will provide the reply to Section C in a supplemental response.

The matrix of Final Safety Analysis Report changes requested by the Staff in Request E.1 is provided by Enclosure 2. The commitments made in this letter are contained in Enclosure 3. If you have any questions, please telephone me at (256) 729-2636.

Sincerely,

Karl W Singer

for C. M. Crane

Enclosures
cc: See Page 4

REFERENCES

1. TVA letter to NRC dated October 1, 1997, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-384 - Request For License Amendment for Power Uprate Operation

2. TVA letter to NRC dated March 16, 1998, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 Technical Specification (TS) No. 384 Supplement 1 - Request for License Amendment for Power Uprate Operation

3. NRC letter to TVA dated March 13, 1998, Browns Ferry Nuclear Plant, Units 2 and 3: Request for Additional Information Relating to Technical Specification Change No. TS-384 - Power Uprate Operation (TAC Nos. M99711 and M99712)

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ENCLOSURE 1
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 2 AND 3

BROWNS FERRY NUCLEAR PLANT (BFN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING UNITS 2 AND 3 TECHNICAL SPECIFICATION (TS) CHANGE TS - 384, REQUEST FOR LICENSE AMENDMENT FOR POWER UPRATE OPERATION (TAC NOS. M99711 AND M99712)

This enclosure provides the TVA response to the March 13, 1998, NRC request for additional information (Reference 6).

A. Spent Fuel Storage

NRC Request A.1

Since the spent fuel pool (SFP) heat loads will increase because of plant operations at the proposed increased power level, provide the following information:

a. Provide/compare the heat loads and corresponding peak calculated SFP temperatures (for plant operations at current power level and at proposed uprate power level) during planned refueling and unplanned full core off-load. A single failure of SFP cooling system need not be assumed for the unplanned full core off-load.

b. Is full core off-load the general practice for planned refuelings?

c. How many SFP cooling system trains will be available/ operable prior to a planned refueling outage or an unplanned full core off-load?

TVA Reply A.1

a. The spent fuel pooling heat load for BFN at the power uprate condition was evaluated based on the applicable BFN fuel design characteristics and 24-month fuel cycle length. This power uprate condition heat load is considered applicable to both normal refueling operation and unplanned full core off-load.

No specific calculations were made for the peak spent fuel pool for normal operation and unplanned full core off-load. The design basis for the fuel pool cooling system remains the same for the pre and post power uprate conditions, e.g., the system is capable to maintain a peak pool temperature below 125°F for normal refueling and below 150°F for an unplanned full core off-load. Unloading the reactor core and the associated increase in fuel pool heat load is a controlled evolution. Administrative controls are used to ensure that the pre-uprate fuel pool heat load does not exceed available cooling capacity, such that the fuel pool gates are not closed until the decay heat load is less than or equal to the fuel pool cooling heat exchanger capacity. These administrative controls are also applicable to the power uprate condition. Therefore, at the power uprate condition, the peak spent fuel pool temperature will also remain below 125°F for normal refueling and below 150°F for an unplanned full core off-load.

b. No, full core off-load is not the general practice for planned refueling outages at BFN. TVA has previously performed full core off-loads and maintains this capability.

c. At BFN, there are two spent fuel pool cooling trains. The residual heat removal (RHR) system in supplemental SFP cooling mode can also be placed in service to provide additional cooling capability. Prior to a planned refueling outage (including full core off-loads) calculations are performed to determine the pool heat load and determine which equipment must be placed in service to maintain pool temperature. As stated in the response to item A.1.a above, there are administrative controls to ensure that the fuel pool cooling capacity is not exceeded during core offload.

NRC Request A.2

Discuss the provisions (actions) established in plant operation procedures to provide the controls necessary to ensure that the limiting condition for operation, LCO 3.10.C.2 temperature limit of 150 degrees F will not be exceeded.

TVA Reply A.2

The spent fuel pool temperature is measured and recorded once a day in accordance with the requirements of Technical Specification Surveillance Requirement 4.10.C.2 in Surveillance Instruction (SI) 2, "Instrument Checks and Observations." This requirement will be transferred to the Technical Requirements Manual (TRM) as part of TVA's move to

Improved Standard Technical Specifications (ISTS). As such, the requirement to verify spent fuel pool temperature will be maintained as a Surveillance Requirement under the same title.

This SI administratively limits the maximum spent fuel pool temperature to 125°F. If the spent fuel pool temperature exceeds 125°F, plant Abnormal Operating Instruction (AOI) 78-1, "Fuel Pool Cleanup System Failure," provides the operator actions that will be taken to maintain the temperature below 125°F.

These actions include correcting any malfunctions of the SFP cooling system, starting the RHR system in the shutdown cooling mode if the fuel pool gates are removed, starting the RHR system in the supplemental fuel pool cooling mode, and, for Unit 2 only, interconnecting the Unit 1 SFP cooling system. The SFP level would be maintained as discussed in the response to request A.3 below.

NRC Request A.3

In the unlikely event that there is a complete loss of SFP cooling capability, the SFP water temperature will rise and eventually will reach boiling temperature. Provide the time to boil (from the pool high temperature alarm caused by loss-of-pool cooling) and the boil-off rate (based on the heat load for the unplanned full core off-load scenario). Also, discuss sources and capacity of make-up water and the methods/systems (indicating system seismic design Category) used to provide the make-up water.

TVA Reply A.3

The spent fuel pool is normally maintained below 125°F as discussed in Section A.2 above. For a complete loss of SFP cooling capability, the time required for the pool temperature to rise from 125°F to 212°F will be approximately 5.6 hours. The boil-off rate has been calculated to be approximately 86 gpm. Approximately 28.9 hours (from the onset of boiling) are available to provide a make-up water supply to the pool prior to reaching the minimum shielding height of 8.5 feet above the top of the spent fuel. In addition to the 5% power uprate effects, this evaluation considers 24-month fuel cycle and the ANSI/ANS 5.1-1979 + 2σ uncertainty decay heat curve.

To assure adequate make-up under a postulated condition (i.e., fuel pool water boil off) the RHR/RHR Service Water crosstie provides a permanently installed Class I seismically qualified make-up water source for the spent fuel pool. The capacity of this system is in excess of 3000 gpm. This ensures that irradiated fuel is maintained submerged in water and that reestablishment of normal fuel pool water level is possible under anticipated conditions. Two additional sources of spent fuel pool water make-up are provided via standpipe and hose connections (150 gpm each) on each of the two class 1 seismically qualified emergency equipment cooling water headers.

NRC Request A.4

The environmental qualification (EQ) of mechanical equipment inside and outside containment has not been addressed. Please demonstrate that plant operations at the proposed uprated power level will have no impact on the EQ of mechanical equipment inside and outside containment.

TVA Reply A.4

Browns Ferry was licensed prior to the establishment of NRC General Design Criteria (GDC) 4, "Environmental and Dynamic Effects Design Bases." Consequently, BFN is not committed to GDC-4 and; therefore, does not have a formal mechanical equipment qualification program. Accordingly, environmental qualification of mechanical equipment inside and outside of containment is not addressed by power uprate.

B. ELECTRICAL POWER, AND AUXILIARY SYSTEMS

NRC Request B.1

With the thermal power uprated from 3293 Mwt to 3458 Mwt at Browns Ferry Nuclear Plant (BFN) Units 2 and 3, provide the net electrical power output increase for each unit resulting from the proposed power uprate. Discuss the potential impact the additional heat has on the main generator and its auxiliary equipment due to power uprate. Specifically, address in this discussion, the main generator stator and rotor, exciter and voltage regulator, hydrogen cooling system, and the generator protective relays.

TVA Reply B.1

The net electrical power output increase for each unit will be approximately 57,500 KW. This output is based on steam conditions of 980 psia and with modified 1^{st} and 2^{nd} stage turbine diaphragms.

The power uprate evaluation of the main generator and its auxiliary equipment indicated that there are no hardware changes or modifications necessary for uprated operation with the operation of Browns Ferry Units 2 and 3 remaining within the original capability curves. The following are specific systems evaluations results:

- Generator Review

  The Browns Ferry generators are designed to operate within the specified temperature rises and in accordance with other standards established in the applicable ANSI and IEEE Standards. Operation in excess of the generator capability curves causes increases in copper temperature, thermal expansion, insulation stresses, and other conditions that are in excess of the criteria specified when the generator was engineered. Operation of the generators beyond loads exhibited on the generator capability curves for the various gas pressures is not recommended or permissible.

  For the power uprate, the steam dome pressure will be increased to 1050 psia. Modifications to the turbine 1st and 2nd stage flow passing capability will also be performed to accommodate the increased steam flow at the power uprate condition. Based on these conditions, turbine heat balance calculations were performed to determine the throttle pressure required to match the uprated reactor thermal power. For the reactor dome pressure of 1050 psia, two uprated steam conditions were theorized. One was based on an estimated steam line pressure drop of 70 psi. This condition corresponds to a turbine inlet pressure of 980 psia. The second condition was based on a potential improved pressure drop of 40 psi (i.e., 30 psi less). This would correspond to a turbine steam inlet pressure of 1010 psia.

  The turbine-generators were evaluated at the highest attainable throttle pressure of 1010 psia and valve wide open (VWO). This operating point is the bounding analysis and would cover the operating condition if the high pressure (HP) turbine section was modified and operated at 1010 psia. Therefore, there will be no structural integrity issue with operating at 1010 psia with a modified HP section. Thermodynamically the control valve will be throttling more at 1010 psia than at 980 psia. There will be some performance loss due to additional throttling. There will be no issue with the control system for the condition of a modified HP section and operation at 1010 psia.

For the proposed uprate of 105% of original core thermal power, the KVA rating of unit's 2 and 3 generators would remain at the original rating of 1,280,000 KVA. At the new valves wide open (VWO) conditions, the expected generator output was calculated to be 1,188,908 KW for 980 psia and 1,216,864 KW for 1010 psia throttle pressure. In order to provide the required generator capability for these ratings, the power factor will be maintained at 0.93 (or 0.95 for 1010 psia pressure) or higher. The generator's components are acceptable for operation at the uprated power level since the units will be operating within the original capability curves.

Because the units will be operated within the original generator capability curves, the change in heat load from the current generator operating condition to the uprate condition is already factored into the design of the generator and its auxiliary systems. They can accommodate the uprate.

- Generator Stator Cooling System

The generator stator cooling water system requirements increase with an uprate because additional cooling water flow is needed due to increased stator bar heating, however, the heat load requirement remains below the system design rating. The maximum rated cooling water design flow requirement for the system does not change. The existing stator water cooling system will be adequate for the proposed uprate with generator operation at the original rating of 1,280,000 KVA and at a power factor of 0.93 or higher. The generator stator water coolers performance was also evaluated and found to be acceptable for the power uprate condition.

- Hydrogen Coolers for Rotor

At the original generator rating of 1,280,000 KVA and at a power factor of 0.93, the existing hydrogen coolers are capable of reliable operation at the uprated heat load. The total water flow requirement to the coolers will remain within the equipment design limit, based on a fouling resistance of 0.001 and a maximum water inlet temperature of 100°F. The heat load on the hydrogen coolers is removed by the Raw Cooling Water (RCW) system. The RCW system performance for this function was evaluated and found to be acceptable at the power uprate conditions.

- High Voltage Bushings and Current Transformers

  The in-service high-voltage bushings and current transformers were reviewed for the power uprate conditions and are concluded to have adequate design margin to handle the associated heat load.

- Stator Winding

  The in-service stator winding assembly has adequate design margin at the uprated conditions. The reactive capability curves remain the same for the stator winding limitations. Thus, there is no adverse impact on heat loads due to operation at the power uprate condition.

- Exciter and Voltage Regulator Cooling Systems

  The excitation system includes the exciter, voltage regulator and rectifier. The excitation system was evaluated for the power uprate condition and its performance associated with the increased heat load was determined to be acceptable. The performance of the exciter cooling system at the power uprate conditions was reviewed and determined to be adequate to handle the associated heat load.

- Generator Protective Relays

  The generator protective relays were evaluated as part of the grid stability at the power uprate condition. The results indicated that the generator protective relays' performance is acceptable for the power uprate condition. There is no adverse impact on the generator breaker cooler performance since the generator operation remains within the pre-uprate design ratings.

NRC Request B.2

Discuss the impact that power uprate has on all levels of the station auxiliary electrical onsite distribution and offsite power system by providing bus voltages and power flow changes from before and after power uprate load flow studies. This discussion should include a specific assessment for the main step-up transformer, startup transformer, unit auxiliary transformer, emergency diesel generators, and the iso-phase buses.

TVA Reply B.2

The method used to analyze the adequacy of the Onsite Power Distribution system was as follows:

(1) Nuclear steam supply system (NSSS) and balance of plant (BOP) system analyses for power uprate were reviewed for changes or additions of electrical loads that would result from operation at power uprate conditions.

(2) TVA electrical calculations were reviewed to determine the existing, pre-power uprate, capacity of each electrical subsystem/component against the design basis. The review of the TVA calculations also determined how the existing electrical system/component was evaluated against the design basis (i.e., if component nameplate data was used or if the actual power requirements at a plant operating condition was used).

(3) The revised power uprate load requirements from the NSSS and BOP system evaluations were then evaluated compared to the originally designed equipment capabilities. This comparison was performed to verify that the equipment as originally designed could perform satisfactorily under the new uprate conditions.

The effect of power uprate on the onsite electrical system is to increase the electrical power requirements of power generation equipment, principally the condensate, condensate booster, and recirculation pumps. The existing, pre-power uprate, electrical analysis typically uses motor nameplate data for evaluating the equipment against design requirements so only changes in electrical loads that exceed the existing component nameplate data would affect the analysis. The only exception to the use of nameplate data was found in the pre-uprate Emergency Diesel Generator Load studies and the 4.16 KV and 480 V Busload and Voltage Drop studies which consider the RHR and core spray (CS) pump loads based on brake horsepower for the RHR and CS system flow requirements during a loss of coolant accident (LOCA). See the response to NRC Request B.4 for further discussion. The pump flow/head requirements for the RHR and CS systems during LOCA were confirmed to not change due to power uprate. Therefore, the basis used in the pre-uprate electrical calculations was still valid for power uprate. At the time of the original submittal of TS-384, the determination of changes caused by power uprate, to the electrical power requirements of motor operated valves included within the scope of Generic Letter (GL) 89-10, had not been concluded. The power uprate electrical system analysis for the TS-384 submittal assumed a conservative doubling of the electrical load for all GL 89-10 valves impacted by power uprate in order to determine if sufficient margin existed in diesel generator capacity. The power uprate electrical analysis concluded that sufficient margin existed in diesel generator capacity and that the assumed additional load due to any GL 89-10 modifications would have an insignificant impact on the station load flows.

As part of the power uprate implementation, all GL 89-10 motor operated valves impacted by power uprate were evaluated. See the response to NRC request B.6. This evaluation has concluded that no increase in motor terminal voltage nor increase in motor horsepower, above the existing pre-uprate valve motor operator capabilities, are required for any GL 89-10 motor operated valves affected by power uprate. With the conclusion of the GL 89-10 evaluation for power uprate, it is concluded that the pre-uprate electrical power load flows, diesel generator load flows, and station battery load flows are within capability for the 5% power uprate.

- Main Step-Up Transformer

  The Browns Ferry Unit 2 and 3 main transformers are rated at 1200 MVA. The main transformer was evaluated against the maximum rated generator power production predicted for the power uprate condition to establish the minimum generator power factor that could be tolerated without exceeding the transformer MVA rating. This evaluation was based on normal electrical line-up with the generator output supplying house loads. At the uprated electrical outputs of 1156 MWe for Unit 2 and 3 with a 0.94 and 0.95 generator power factor respectively for Unit 2 and 3, the transformer loads for both configurations are less than the 1200 MVA rating.

- ISO Phase Bus

  The ISO phase bus cooling systems for Browns Ferry Units 2 and 3 were evaluated at current levels, 32,376 Amps, and for the proposed uprated condition of 33,591 Amps. These units are rated for 35,270 Amps (16,500 Amps self-cooled) and, therefore, are still below the bus duct current rating. A review of the layout drawings showed that the present arrangement is able to carry the expected 33,591 Amps, which is below the maximum amperage rating of 35,270.

NRC Request B.3

State whether the added generation will affect the offsite system grid voltage profiles and system grid stability. With the added generation, does BFNP require re-analysis of Branch Technical Position PSB-1 and changes in the degraded grid setpoints? Provide summaries of the grid stability cases reviewed and attendant findings.

TVA Reply B.3

Offsite system grid profiles and system stability studies have been performed for BFN at the uprate conditions. The analysis concluded that during lightly loaded conditions, the stability margins for BFN generators is reduced because there is less load for damping power swings as a result of system disturbances. The new stability limits for BFN Units 2 and 3 have been addressed in an operations Standing Order. PSB-1 does not require re-analysis since the methodology and software have not changed. The degraded grid setpoint is not affected by the added generation because there are no load changes for the safety related power distribution system except for GL 89-10 valves as stated in reply to NRC Request B.2.

NRC Request B.4

Explain why station loads under normal and emergency operational conditions are computed using equipment nameplate data except for the core spray and RHR pump motors where the actual brake horsepower for the flow conditions is used.

TVA Reply B.4

Equipment nameplate data is typically used in power load studies because it provides the most conservative results (i.e., lowest margin) in the electrical analysis. This conservative methodology, therefore, alleviates some programmatic requirements to periodically monitor all station electrical loads to verify the results of the electrical load calculations. The calculations would therefore only need revision if operational capabilities require a modification of an electrical device to one of higher than the original nameplate capabilities. However, there are conditions in which the use of nameplate data is so overly conservative, that it would necessitate excessive over-sizing of power supply or power distribution equipment.

In the case of the CS and RHR pumps, use of nameplate data would be overly conservative, specifically when considering how the equipment actually operates. In case of automatic actuation of the CS and RHR pumps, the pumps start in min-flow mode and then revert to reactor vessel injection mode once the system injection valves open. The actual motor horsepower requirements are significantly less than the motor capability. The actual motor requirements determined from brake horse-power to flow conditions does not result in incorrectly calculating equipment loading. As stated earlier, the pump flow/head requirements for the RHR and CS systems during LOCA were confirmed to not change due to power uprate.

NRC Request B.5

As a result of lessons learned from the Main Yankee Independent Safety Assessment Inspection, all licensees are required to review and evaluate whether the power uprate would alter the original licensing basis of General Design Criterion (GDC)-17 and station backout (SBO) requirements. Please provide BFNP's assessment regarding GDC-17 and SBO requirements.

TVA Reply B.5

GDC-17, "Electric Power Systems" provides the requirements for onsite and offsite power system safety functions. Safety functions of the offsite and onsite electric power systems are not impacted by uprated conditions.

In general, the electrical power system safety function capabilities do not change as a result of power uprate unless there are changes in electrical components such that loading on the onsite power sources are increased. While power uprate will increase the core decay heat load, this will only require longer operation of the systems needed to bring the plant in a cold shutdown condition. These systems do not require increased capability to cope with the decay heat associated with power uprate. There is no increase in the on-site power sources loading associated with power uprate operation, therefore, it can be concluded that power uprate has no impact on GDC-17 requirements.

Browns Ferry response to a postulated SBO event is to utilize the reactor core isolation cooling (RCIC) system for reactor shutdown. Cooling of the control room and equipment rooms is augmented from excess diesel capacity associated with the non-blacked out units to cope for 4 hours. A controlled depressurization to about 200 psig is initiated within ten minutes of the SBO.

The power uprate impacts the decay heat and reactor temperatures. This impacts the heatup of equipment spaces and the consumption of cooling water during the required coping duration. TVA has confirmed that system limits are not impacted by the increases in decay heat or reactor pressure.

NRC Request B.6

In the TVA submittal for Generic Letter 89-10, it is noted that BFNP may require replacement of some valve motor operators which may add small additional loads to the emergency diesel generators. BFNP has committed to verify the emergency diesel generator capacity. Provide the findings for this commitment.

TVA Reply B.6

The results of the GL 89-10 program evaluation showed that there are no motor changes required and that the replacement of one torque switch and reset of three other torque switches will be needed for power uprate implementation. There are no electrical equipment changes associated with these modifications. Therefore, there is no impact on the emergency diesel generator loading resulting from the GL 89-10 evaluations for power uprate.

C. ENVIRONMENTAL QUALIFICATION FOR SAFETY-RELATED ELECTRICAL EQUIPMENT

NRC Request C

For each component/equipment type (or one representative/ bounding example of a component/equipment type) where expected environmental conditions at the uprate power level exceeds the environmental conditions tested to, provide the following:

1. A description showing the relationship between environmental conditions (i.e., temperature vs. time) tested to, the expected environmental conditions at current power levels, and the expected environmental conditions at the power uprate level from time 0 (initiation of accident) to the time the component/equipment type is required to remain operable for post LOCA operation.

2. An evaluation demonstrating qualification for each segment of the uprate power level temperature response that is not

enveloped by the environmental conditions (i.e., temperature) tested to.

3. Where (or if) margins derived through the use of the Arrhenius methodology are utilized as part of the basis for concluding continued qualification, provide the Arrhenius calculation at the current (if applicable/available) and uprate power levels. Define the margins available for the current and uprate power levels and describe and justify the reduced margin for the uprate power level.

TVA Reply C

The post-accident environmental evaluations for uprated conditions have not been completed. However, preliminary evaluations have been performed for equipment located inside the drywell. These preliminary evaluations indicate the drywell accident profile for uprated conditions are enveloped by the equipment test profiles. Due to conservatism in the current drywell accident profiles, preliminary calculations performed using the Arrhenius methodology indicate there will be no reduction in margin in the accident degradation equivalency comparisons performed at uprated power conditions. TVA will provide the results of this effort and a reply to this section when completed.

D. REACTOR AND REACTOR SYSTEMS

NRC Request D.1

The standby liquid control (SLC) system pump discharge pressure is increased by 50 psig (Page E1-6, Enclosure 1 to TVA letter dated October 1, 1997), but the increase in reactor operating pressure due to power uprate is only 30 psig. Allowances for system test inaccuracies were supposed to be in the original values. What is the basis for the 50 psig increase?

TVA Reply D.1

As stated in Enclosure 5, Section 6.5 from the October 1, 1997 submittal (Reference 1), the SLC system surveillance test pressure is increased by 50 psi, from a pre-uprate value of 1275 psig to 1325 psig at the power uprate condition. The test pressure is based on the lowest Main Steam Relief Valve (MSRV) opening setpoint, including allowance for drift.

At the pre-uprate condition, the lowest MSRV opening setpoint is 1105 psig + 1%, or 1116 psig. A request for MSRV setpoint tolerance relaxation to +3% has been previously submitted to the NRC in TS-386 (Reference 8). The operating dome pressure will be increased by 30 psi for the power uprate condition. However, for the purpose of this analysis, the MSRV opening setpoints are conservatively assumed to increase by 35 psi along with a 3% setpoint tolerance, yielding the lowest MSRV opening setpoint value of 1140 psig + 3%, or 1174 psig. The net increase in the MSRV opening setpoint from pre-uprate to uprate conditions is 58 psi.

Based on this MSRV opening setpoint increase, the pre-uprate SLC surveillance test pressure of 1275 psig is increased by 50 psi to 1325 psig. This is a round-off increase which is within the 25 psi uncertainty allowable for this setpoint.

NRC Request D.2

It is stated (Page E1-8, Item # 5, Enclosure 1 to TVA letter dated October 1, 1997) that "Due to human factors consideration, the value of 1175 psig was chosen." The value of 1175 psig was chosen instead of 1176.5. Why wasn't the value of 1176 or 1177 chosen instead of 1175?

TVA Reply D.2

The value of 1175 psig is chosen instead of 1176 or 1177 to ease the operator recognition and understanding that automatic protective action is about to occur.

NRC Request D.3

Why are the maximum operating pressures for high-pressure coolant injection (HPCI) and reactor core isolation coolant increased by 54 psi and not 30 psig (Page E1-11, Item # 10, Enclosure 1 to TVA letter dated October 1, 1997)?

TVA Reply D.3

The most significant impact of power uprate on the HPCI system is the higher reactor operating pressure and the corresponding increase in MSRV setpoints. The HPCI system was originally designed to provide injection into the reactor pressure vessel to a reactor pressure of 1120 psig. For system and accident analyses at the power uprate condition, the MSRV opening setpoint is conservatively increased by 35 psi to 1140 psig to ensure that adequate MSRV simmer margin is provided. It is noted that the actual pressure will increase only 30 psi rather than the 35 psi assumed in the analysis. In addition, an increase in the setpoint tolerance from 1% to 3% is also being evaluated as part of the valve setpoint relaxation program. The combining of these two changes increases the maximum HPCI system injection pressure to 1174 psig (1140 psig + 3%), for a 54 psi increase in pressure.

NRC Request D.4

It is stated (NEDC-32751P, Section 1.2.1, Enclosure 5 to TVA letter dated October 1, 1997) that some analyses are performed at 100% rated power. Identify the portions of the analysis where the power levels were assumed at 100 percent power and explain why analyzing at 100% power is acceptable for the uprated conditions.

TVA Reply D.4

Analyses of the limiting fuel thermal margin transient events, such as Generator Load Rejection with No Bypass, Turbine Trip with No Bypass and Feedwater Controller Failure Maximum Demand, are performed at 100% licensed power. These events are analyzed with the ODYN code and GEMINI methodology which include allowance for core thermal power uncertainty. The GEMINI methodology is documented in the NRC approved GESTAR document. The application of GEMINI methodology for power uprate transient analyses is also documented in Appendix E of NEDC-31897P-A "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," May 1992.

NRC Request D.5

It is stated (see NEDC-32751P, Section 2.4, *Stability*, Enclosure 5 to TVA letter dated October 1, 1997) that the BFNP will rely on the revised interim corrective actions for both units until the LTS Option III is implemented. What is the schedule for implementing Option III?

TVA Reply D.5

In a letter dated July 10, 1996 (Reference 7), TVA provided the installation schedule for the Option III methodology of the advanced digital power range neutron monitoring (PRNM) system as requested by GL 94-02. In that letter, TVA indicated that an upgrade to PRNM system would occur on Unit 2 during the cycle 10 refueling outage, and Unit 3 during the cycle 8 refueling outage. For each unit, TVA will operate the PRNM upgrade with the Option III trip in the "indicate only" mode during the first cycle of operation. Following that cycle of operation, TVA intends to enable the stability trip function. That is, TVA intends to enable the trip function on the Unit 2 PRNM system prior to startup from the cycle 11 refueling outage, and Unit 3 prior to startup from the cycle 9 refueling outage.

NRC Request D.6

Confirm that credit is not taken for the relief flow (see NEDC-32751P, Section 3.4, *Reactor Overpressure Protection*, Enclosure 5, to TVA letter dated October 1, 1997). Also, specify the safety/relief valve set points used in the analysis. Identify the NRC-approved model used in the analysis.

TVA Reply D.6

The Reactor Vessel Overpressure Protection analysis only takes credit for the MSRVs operation under spring set pressure. The externally actuated mode, via electro-pneumatic actuator, is not assumed in this analysis. The power uprate overpressure analysis conservatively assumed a 35 psi increase in the MSRV opening setpoint to bound the 30 psi dome pressure increase as shown in the license amendment request (see Table 5-1 of Enclosure 5 to Reference 1). At the pre-uprate condition, the technical specifications value for the MSRV setpoint tolerance is ±1%, and the corresponding analytical value is ±3%. The power uprate analysis also assumed ±3% for MSRV setpoint tolerance. The analytical setpoints used for power uprate analysis are shown below. For comparison purposes, the pre-uprate values are also shown.

Pre-Uprate	Power Uprate
1105 psig + 3%	1140 psig + 3%
1115 psig + 3%	1150 psig + 3%
1125 psig + 3%	1160 psig + 3%

The vessel overpressure protection analysis is based on the limiting event Main Steam Isolation Valve (MSIV) closure with Flux Scram (failure of MSIV position scram). This event is analyzed using the ODYN code consistent with the NRC approved GESTAR methodology.

NRC Request D.7

Higher pump speed is expected (Section 3.4, *Reactor Recirculation System*, Enclosure 5, to TVA letter dated October 1, 1997), but it is not clear how much increase is expected. Specify the expected increase in speed. Describe the plant experience with higher pump speed operation such as increased core flow and pump vibration problems, if any.

TVA Reply D.7

The expected increase in pump speed at uprated plant conditions at 100% core flow is 20 rpm. This represents an increase in speed of 1.3% based on an increase from 1584 rpm at pre-uprate conditions to a final pump speed of 1604 rpm at uprated conditions.

Even though BFN is licensed for Increased Core Flow (ICF) operation, TVA does not typically utilize ICF as part of the plant operational strategy and therefore has not compiled any substantial history involving operation at higher pump speeds. Therefore, TVA's experience with higher pump speed and/or vibration problems resulting from higher speeds is limited. The increase in pump speed for ICF conditions places the reactor recirculation system (RRS) at the upper region of its design performance conditions. Operational limitations involving higher recirculation flow and/or vibration will be documented and dispositioned. Vibration monitoring is provided on Units 2 and 3 for the recirculation pump motor, pump shaft, and pump case.

NRC Request D.8

In NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Reactor," May 1992, the staff required that plant-specific submittals must address the modifications described in General Electric (GE) Service Information Letter (SIL) No. 377. Section 3.8, *Reactor Core Isolation Cooling*, Enclosure 5, to TVA letter dated October 1, 1997, does not address the SIL 377 modifications.

TVA Reply D.8

The turbine control system modifications described in GE SIL 377 restricts steamflow to the turbine until after the turbine is under governor valve control, thereby minimizing the transient peaks in turbine speed, pump discharge pressure and flow during a quick start. This SIL identifies modifications primarily intended for the larger GS-2 model turbine. Although the same modification would dampen the startup transient observed in the smaller GS-1 turbine used in the Browns Ferry plants, operating experience with the GS-1 turbine indicates that it is not as susceptible to a transient overspeed conditions during a quick start. The increase in the maximum RCIC system operating pressure resulting from power uprate is not expected to result in transient speed peaks that would require a modification similar to that described in SIL 377. Therefore, BFN does not plan to implement the modifications described in SIL 377. Service Information Letters are issued for information purposes only and do not impose new requirements for system modifications. Power uprate does not change the recommendations presented in SIL 377.

NRC Request D.9

Confirm that the reliability of the HPCI System will be monitored in accordance with the criteria which might have been developed to comply with the maintenance rule 10 CFR 50.65 (Section 4.2.1, *High Pressure Coolant Injection (HPCI)*), Enclosure 5, to TVA letter dated October 1, 1997).

TVA Reply D.9

The HPCI system is currently within the scope of the Maintenance Rule for Units 2 and 3 at Browns Ferry. The HPCI system is categorized as risk-significant, standby equipment, and is monitored using unavailability and reliability performance data. Monitoring of HPCI performance in accordance with 10 CFR 50.65 will continue at uprated conditions.

NRC Request D.10

Analysis power assumed for the GEMINI analyses is only 100% (Table 9-1, *Parameters used for Transient Analysis*, Enclosure 5, to TVA letter dated October 1, 1997). Justify the analyses at 100 % instead of 102%. Also, the steam flow for the power uprate analysis is assumed at 100% instead of 106%. Justify the analysis with only 100% steam flow.

TVA Reply D.10

Transients analyses using the GEMINI methodology are modeled with the ODYN computer code. For those events which are analyzed for fuel thermal limits consideration (such as Generator Load Rejection, Turbine Trip and Feedwater Controller Failure), the initial core thermal power is assumed at 100% rated power. The exception is for the vessel overpressure conformance analysis which based on the limiting event MSIV Closure with Flux Scram (failure of position scram). This MSIV Flux Scram event is analyzed with the ODYN code at 102% of rated power to ensure that the maximum peak vessel bottom pressure is obtained.

For transients events using the REDY code, such as Inadvertent HPCI Actuation, the initial core thermal power will be 102% of rated, as required by the REDY code application.

The ODYN and REDY computer models and the GEMINI methodology have been approved by the NRC for use in boiling water reactor (BWR) transients reload licensing analyses and documented in GESTAR. See also the response to NRC Request D.4 regarding the use of 100% core thermal power for transients analyses.

The 106% steam flow corresponds to the BFN original design basis transient analysis requirement of 105% core thermal power as initial condition. As stated in the above paragraph and also in the response to NRC Request D.4, BFN current transient analysis is based on the NRC-approved GESTAR document. As such, the ODYN code and GEMINI methodology change the transient analytical basis to 100% power for initial conditions. Thus, the vessel steam flow of 14.14 Mlb/hr, as shown in Table 9-1 of Enclosure 5 to Reference 1, represents the steam flow corresponding to 100% uprated power.

NRC Request D.11

Please refer to Section 10.5, *Required Testing,* Enclosure 5, to TVA letter dated October 1, 1997 (Reference 1).

a. Tests will be required on the recirculation system to demonstrate flow control over the entire pump speed range to enable a complete calibration of the flow control instrumentation. These tests should also assure that no undue vibration occurs at uprate conditions.

b. Startup tests on HPCI during the initial startup after being licensed at uprated power will be required.

TVA Reply D.11

a. Testing for the RRS is described below.

   RRS Control System Testing

   The predicted increase in RRS drive flow, an increase in RRS pump speed of approximately 20 rpm, will be necessary to maintain the rated core flow. Uprated power level is within the operating conditions already demonstrated. Furthermore, BFN limits operation of the RRS flow control system to the lowest manual mode, hence there is, no auto control feedback. Therefore, no special RRS testing in addition to the normal testing performed as part of the

Refueling Test Program is planned for the implementation of power uprate.

Core Flow Calibration

The calibration of the total core flow measurement instrumentation is normally performed as part of the Refueling Test Program and is not related to the power uprate program implementation.

Vibration

Power uprate does not increase the core flow of the BFN units which have been previously licensed for ICF to allow additional operating flexibility. GE SIL 600 has concluded that the vibration/noise encountered at another BWR implementing power uprate (NRC Information Notice 95-16, "Vibration Caused By Increased Recirculation Flow in a Boiling Water Reactor") was not attributable to power uprate. In addition, based upon evaluations performed by GE for similar plants, the BFN RRS piping is judged to be acceptable for flow induced vibrations due to 105% power uprate conditions with no vibration monitoring requirements. Therefore, no special vibration monitoring of RRS components is planned as part of power uprate implementation. In the event that BFN operates in the ICF region, the increase in pump speed for ICF conditions paces the RRS at the upper region of its design performance conditions. Operational limitations involving higher recirculation flow and/or vibration will be documented and dispositioned. Refer also to response to Item D.7.

b. System testing is currently being evaluated to determine the startup/surveillance testing that is required to demonstrate the ability of plant systems to perform their designed safety functions. HPCI is included in this evaluation. Startup/surveillance testing will be conducted to demonstrate the ability of plant systems to perform their designed functions under uprated conditions as defined by the start-up test program.[1]

NRC Request D.12

As a result of power uprate, a number of variables and limits utilized in the Emergency Operating Procedures (EOP) may be affected (Section 11.1.2.3, *Emergency Operating Instructions*,

[1] This commitment was previously made in Reference 1.

Enclosure 5, to TVA letter dated October 1, 1997). GE report NEDC-32751P states that "The plant EOIs will be reviewed for any effects of power uprate, and the EOIs will be updated as necessary." Confirm that TVA performed a review of the EOP variables and limit curves for the uprate conditions.

TVA Reply D.12

For BFN, the Emergency Operating Procedures are designated as Emergency Operating Instructions (EOI). Review and revision of the EOIs, which includes a review of all EOI variables and limit curves, for changes due to power uprate is an ongoing activity during the power uprate implementation phase. The review, update of the EOIs for any changes, and training are scheduled to be completed prior to the power uprate start-up for Browns Ferry Unit 3.

NRC Request D.13

The following items in the acceptance criteria are not addressed: fuel integrity, radiological consequences, containment pressure, reactor oscillations and long term shutdown and cooling (Section 9.3.1, ATWS, Enclosure 5, to TVA letter dated October 1, 1997). What is the calculated peak containment pressure?

TVA Reply D.13

Section 9.3.1 of Enclosure 5 showed that for an anticipated transient without scram (ATWS) condition, the resulting fuel peak clad temperature is 1499°F and the peak suppression pool bulk temperature is 190°F. These results are within the BFN design criteria, namely 2200°F for peak clad temperature and 281°F for wetwell shell design temperature. The peak containment pressure during an ATWS condition is calculated at 11 psig, and still within the BFN design pressure of 56 psig.

The radiological consequences and the long-term shutdown and cooling capability were previously addressed in a BWR Owners Group study on the thermal-hydraulic instability in a BWR associated with an ATWS event ("ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," NEDO-32047, February 1992). This evaluation provides a bounding set of expected system responses relative to the BWR fleet. The results demonstrate that the potential for core thermal-hydraulic oscillations during an ATWS event:

(1) is not expected to result in any significant core distortion (i.e., would impede core cooling, prevent safe shutdown, or threaten primary system integrity)

(2) presents no additional threat to the primary system integrity, containment or long-term cooling, and

(3) does not significantly increase the radiological consequences, which remain within 10 CFR 100 limits.

This study was submitted to the NRC by the BWR Owners' Group via letter BWROG-92004, R. D. Binz to A. C. Thadani "ATWS Rule Issues Related to Core Thermal-Hydraulic Stability." As stated in the transmittal letter, specific NRC approval is not requested since no change to the requirements of 10 CFR 50.62 is being proposed. The study is applicable to the BFN power uprate implementation because it assumed the same maximum rod line configuration as for the BFN power uprate operating map. The initial core thermal power is not an important parameter since the key attribute for ATWS events is the runback along the operating rod line toward the natural circulation state point. The radiological consequences in the study are based on a BWR/6 Mark III with open suppression pool and are conservative when applied to the BWR/4 Mark I closed suppression pool since there would be a less direct pathway for the fission products release.

NRC Request D.14

Class 1E battery capacity and compressed air system are not addressed for the scoping analysis (Section 9.3.2, *Station Blackout*, Enclosure 5, to TVA letter dated October 1, 1997).

TVA Reply D.14

TVA has evaluated its batteries during the SBO event and demonstrated a capacity that exceeds 4 hours. Slight differences in the timing of battery loading and certain motor operated valve loads may occur due to higher decay heat and higher reactor operating temperatures with uprated conditions. However, these changes do not significantly impact the ability of the batteries to last 4 hours during the SBO.

Nitrogen usage by the MSRVs may be relied upon for the initial SBO response and subsequent reactor pressure vessel (RPV) depressurization. Slightly higher usage of nitrogen may occur during the first hour of the SBO due to the higher decay heat load and different nitrogen usage also may occur during depressurization. However, overpressure protection may rely upon the MSRV function during the SBO, if necessary, and the HPCI system recirculation mode provides sufficient turbine steam flow capacity to depressurize the RPV without using the MSRVs. Therefore, compressed air and nitrogen usage are not essential for the SBO response with or without uprated power. No action is necessary based on the power uprate.

NRC Request D.15

The staff stated In its Safety Evaluation (SE) for NEDC-31984P (letter W. T. Russell to P. W. Marriott, July 31, 1992) that "Individual licensees should adhere to existing radial power shape limitations when designing core reloads for uprated conditions." Confirm that this requirement is followed.

TVA Reply D.15

The Staff requirement that the power uprate core reload design adhere to existing power shape limitations is intended to assure that the SAFER/GESTR-LOCA emergency core cooling system (ECCS) performance analysis assumption of Counter Current Flow Limiting (CCFL) breakdown in peripheral bundles will remain valid for power uprate conditions. The effect of radial power distribution on CCFL breakdown in the peripheral bundles was generically evaluated for extended power uprate (up to 20% uprated power) in Section 3.3.1 of ELTR2 (NEDC-32523P, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," March 1996), previously submitted and under review by the NRC for approval. The results of that evaluation showed that the radial power shape had an insignificant (<2°F) effect on the peak clad temperature. This evaluation is bounding for the 5% power uprate at Brown's Ferry.

As stated in both NEDC-31984P and the corresponding NRC Safety Evaluation (letter W. T. Russell to P. W. Marriott, July 31, 1992), no change is required to the basic fuel design to achieve the uprated power level or to maintain the safety margins. There is no increase to the allowable peak bundle power. 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 power uprate level. The BFN power uprate submittal has confirmed the acceptability of these limits as determined for power uprate conditions. The plant-specific reload analyses will continue to meet acceptable NRC criteria as specified in GESTAR.

E. MAINE YANKEE LESSONS LEARNED

NRC Request E.1

The submittal included proposed changes to the technical specifications. However, the submittal did not provide any matrix or plan indicating which sections of the Final Safety Analysis Report (FSAR) will be superseded by current extended power uprate re-analysis. Provide a list or matrix that identifies which subsections of the FSAR will be superseded and identify the corresponding sections of the current submittal. The actual updating of the FSAR will be governed by the current regulations, and the affected FSAR subsections should be documented.

TVA Reply E.1

TVA recognizes the importance of accurately and comprehensively updating the FSAR to reflect the changes created by the transition to Uprated Power. A detailed plan is in place and is progressing to evaluate all FSAR sections for power uprate impact. Please find a copy of a matrix (Enclosure 2) identifying sections of the FSAR that are either currently under evaluation for change or where changes are anticipated. It should be noted that only after issuance, by the staff, of the Safety Evaluation Report (SER) for the submitted Technical Specification (TS) Change TS-384 can this task be finalized. The FSAR amendment will be submitted in accordance with current regulations.

The attached matrix is a dynamic vehicle and only represents a "point in time" perspective on the FSAR changes at the time of this letter.

NRC Request E.2

Provide a list of all the computer codes used to perform the re-analysis and indicate if the particular code was approved for the specific application. Respond to the following requests which pertain to the codes used in the power uprate.

a. Review the approving SE for the each code and state whether your application of the code complies with any limitations, restrictions or conditions specified in the approving SE. Demonstrate that your applications of the computer codes in the reanalysis conforms with all assumptions and restrictions given by the corresponding approving SE.

b. In addition, review the SEs for the extended power uprate generic reports and indicate if you complied with all restrictions stated in the approving SE.

TVA Reply E.2

The list of computer codes used in the BFN power uprate safety analyses is provided in the following table:

Name	Description
BILBO	Power/flow operating map and steady-state reactor recirculation system performance evaluation.
ISCOR	Steady-state reactor core coolant hydraulic and internal components ΔPs
PANACEA	3-D core simulator for core-wide transients
ODYN	1-D model for core wide transients analyses
REDY	Point model for core wide transients analyses
SCAT and TASC	Hot bundle CPR calculation
LAMB	Short-term LOCA vessel blowdown flow rate and reactor internal ΔPs
SAFER	Long-term reactor response for postulated LOCA conditions over a spectrum of break sizes and locations.
SAFER/GESTR-LOCA	LOCA fuel peak clad temperature
M3CPT	Short-term containment LOCA loads
SHEX	Long-term containment pressure/temperature
RELAP5	Mass and energy releases for High Energy Line Break (HELB) outside of primary containment
GOTHIC	Reactor building pressure and temperature responses for HELB outside of primary containment
COSMOS/M	Finite element model for main steam thermowells structural evaluation

a. With the exception of RELAP5, GOTHIC, and COSMO/M which are non-GE computer codes, the computer codes used in BFN power uprate analytical process are consistent with those used in previous GE BWR power uprate projects and approved by the NRC for power uprate application in NEDC-31897P-A, "Generic Guidelines for GE BWR Power Uprate," May 1992. The BFN power uprate analytical assumptions include a 5% increase in core thermal power and 35 psi increase in the dome pressure and pressure-related system setpoints. These assumptions are within the boundaries of NEDC-31897-P-A. The BILBO computer is not specifically mentioned in NEDC-31897-P-A, however, this is the same computer code that GE has used in the past to develop all the power/flow map for plant-specific FSARs and power uprate applications. The BILBO computer code is also used in the NRC approved generic power uprate evaluations (NEDC-31984P) and previous plant-specific power uprate applications. The use of the BILBO computer code in the power uprate project conforms with the assumptions and restrictions applied to previous similar analyses.

The computer codes used for ECCS-LOCA performance analyses (LAMB, SCAT, SAFER and SAFER/GESTR-LOCA) have been approved by the NRC for use in power uprate application in NEDC-31897P-A, Appendix D. The ECCS-LOCA computer codes applications are also provided in the "GE Standard Application for Reactor Fuel (GESTAR)," NEDC-24011-P-A-13-US, which is also approved by the NRC. The BFN ECCS-LOCA analyses conform with the review criteria and initial assumptions approved for this code usage in the corresponding SE.

The computer codes used for transients analyses (REDY, ODYN, PANACEA, SCAT, TASC) have been approved by the NRC for use in power uprate application in NEDC-31897-P-A, Appendix E. The ISCOR code is used to generate thermal-hydraulic inputs to the transient codes. The application of ISCOR is provided in GESTAR which is also approved by the NRC. The BFN transients analyses conform with the review criteria and initial assumptions approved for this code usage in the corresponding SE.

The computer codes used for containment short-term and long-term evaluations (SHEX, LAMB, M3CPT) have been approved by the NRC for use in power uprate application in NEDC-31897-P-A, Appendix G. Since SHEX is used for the first time at BFN, a baseline case was performed at the pre-uprate condition in addition to the power uprate condition. The BFN containment analyses conform with the methods and assumptions approved for this code usage.

In the Environmental Qualification area, the HELB analyses are performed using the computer code RELAP 5 for the mass and energy releases outside of the primary containment and the computer code GOTHIC for the reactor building pressure and temperature responses.

The HELB analyses were redone to account for the changes in primary system pressure and temperature associated with power uprate. Mass and energy releases from postulated breaks were generated using the RELAP5 Mod 3 computer code. Temperatures and pressures in the reactor building that result from HELBs were determined using the GOTHIC version 5.0c computer code. The original Browns Ferry Equipment Qualification analyses were performed using RELAP5 Mod 2 and MONSTER. The change in computer codes used was necessitated by a change in the computer hardware used by TVA since the original analyses were performed.

The GOTHIC computer code was written by Numerical Applications Incorporated (NAI). In recent years, a number of utilities, including TVA, in conjunction with EPRI and NAI had developed the GOTHIC computer code to perform a wide range of analyses for determining the pressure and temperature response of buildings due to pipe breaks or perturbations in ventilation systems. GOTHIC represents an advancement in containment and high energy line break analysis over computer codes such as COMPARE and MONSTER. GOTHIC can model three dimensions and includes buoyancy flow in all modeling options. Computer codes of MONSTER's generation do not include buoyancy. For the Browns Ferry power uprate analyses, the GOTHIC options were set up to match the previous analyses that were performed using MONSTER to the extent possible. The analysis used a one dimensional multi-node model and the conditions in each node were assumed to be homogeneous. The only significant difference in the two codes as set up for these analyses is the buoyancy flow model in GOTHIC. The MONSTER reactor building model used for the equipment qualification analysis was converted to be compatible with GOTHIC. The basic model and input assumptions were not changed during the model conversion. Changes were made only as necessary to be consistent with GOTHIC input requirements. These changes did not impact the results. During checkout of the GOTHIC analysis for BFN several analyses was performed to determine if any model changes had been made that would affect the results. This run was made with a test version of GOTHIC in which the buoyancy term was eliminated. The comparison of the results of this run with the MONSTER results showed excellent agreement. Neither code predicted that choked flow would occur in the junctions due to the low differential pressures between nodes. The results obtained for power uprate as described in this submittal are consistent with the physical processes and the results from previous analyses. It is concluded that GOTHIC is a suitable replacement for MONSTER for these types of analyses.

The RELAP5 code has been an NRC and industry standard for a number of years for modeling the thermal hydraulic response of high energy systems including the reactor coolant system. The code was developed to analyze a variety of postulated accidents and transients including a spectrum of pipe break sizes. RELAP5 Mod 3 was used to determine the mass and energy release from various pipe breaks to be input into the GOTHIC model. The models are the same ones that were developed for the original analyses that were performed with RELAP5 Mod 2. There are some differences in the input requirements between RELAP5 Mod 2 and Mod 3. The base RELAP5 Mod 2 models were modified to account for these differences and the change in the reactor conditions associated with increasing the plants power output. RELAP5 Mod 3 is an accepted industry standard for this type of analysis and is an acceptable replacement for RELAP5 Mod 2.

COSMO/M is a finite element computer program developed and maintained by Structural Research and Analysis Corporation. This computer code has been used extensively in many other nuclear power plants. For the TVA power uprate application, this computer code is also benchmarked against classical problems solutions to confirm its accuracy. The power uprate finite element analysis conforms with the assumptions and restrictions used in previous similar analytical applications.

b. The generic guidelines for GE BWR power uprate program (NEDC-31897-P-A, May 1992) have been reviewed and approved by the NRC (Reference letter W. T. Russell to P. W. Marriott "Staff Position Concerning GE BWR Power Uprate Program (TAC No. 79384)," September 30, 1991). In addition, the power uprate generic bounding analyses and equipment evaluations (NEDC-31984P) have also been reviewed and approved by the NRC (Reference letter W. T. Russell to P. W. Marriott "Staff Safety Evaluation of GE BWR Generic Analyses (TAC No. M81253)," July 31, 1992). Both of these documents were reviewed to confirm their application to the BFN power uprate program.

- Review of NEDC-31897-P-A

  The BFN power uprate safety analyses conform with the guidelines as set forth in the Staff position paper. The licensing approach and criteria for 5% power uprate as well as the specific assumptions and bases for power uprate operating condition are consistent with the Staff position paper. In reference to Section 2.2 of the Staff position paper, there are no first-time application of special operational features, such as ICF and Maximum Extended Load Line Limit, combined with the power uprate submittal.

As stated in the response to item E.2.a, the specific assumptions and bases for ECCS-LOCA, transients and containment evaluations are in compliance with the Staff position paper. The analyses boundaries and assumptions are consistent with the guidelines.

Since the control instrumentation and setpoints evaluation is performed using the TVA setpoint methodology, this evaluation was done on a plant-specific basis as indicated by the Staff position paper.

The scope, assumptions and methodology used for the radiological evaluations are consistent with Appendix H of NEDC-31897P-A and the Staff position paper.

The methods and assumptions used for the reactor vessel and internal components evaluations are consistent with Appendix I of NEDC-31897P-A and with the Staff position paper. Reactor water level variation is one of the key parameters considered in the steam dryer/separator performance evaluation. Components' primary stresses evaluations are evaluated against the plant design bases values.

The system equipment evaluations are performed according to Appendix J of NEDC-31897-P-A and include the evaluation of the HPCI and RCIC systems as indicated in the Staff position paper.

The methods and assumptions of the piping evaluations are also consistent with the Staff position paper and include feedwater lines, spargers, certain BOP piping and effects from high energy line breaks (pipe whip and jet impingement).

- Review of NEDC-31894P and Corresponding NRC Safety Evaluation

The Staff SE of the GE BWR Power Uprate Generic Analyses (NEDC-31894P) was reviewed to confirm that the BFN power uprate safety analyses include those which were

identified as required to supplement the generic analyses.

The Loss of Feedwater Flow transient was performed on a plant-specific basis and documented in Section 9.1 and Table 9-2 of Enclosure 5 to Reference 1. The power uprate results concluded that there is no impact on operator actions and response times for this event.

The core thermal-hydraulic stability for BFN at the power uprate condition is addressed in Section 2.4 of Enclosure 5 to Reference 1. BFN plans to implement the Long-Term Solution (LTS) Option III for both Unit 2 and 3; however, the plant will rely on the Interim Corrective Actions (ICA) for both units until the LTS Option III is implemented. The ICA include operational guidelines as described in GE SIL 380 Revision 1 and NRC Bulletin 88-07 Supplement 1.

The conformance to radial power shape for ECCS-LOCA consideration is addressed in the response to Item D.15.

The reload licensing analyses for BFN at the power uprate condition will remain consistent with the methodology shown in NEDE-24011-P-A, latest US revision and supplement.

The Containment Atmosphere Dilution System performance at the power uprate condition is addressed in Section 4.1.4 of Enclosure 5 of Reference 1 on a plant-specific basis. The evaluation concluded that the system is capable to perform its intended function at the power uprate condition and that there is no change to the system operation outside of previously established designs.

In the area of materials and coolant chemistry, TVA will continue to meet the commitments made in the response to GL 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," as required by the Staff SE.

The RHR and CS systems were evaluated on a plant-specific basis and documented in Section 4.2.2 and 4.2.3 of Enclosure 5 to Reference 1, respectively. The results confirm the hardware capability to perform their intended functions at the power uprate condition.

The RCIC and HPCI system performance plant-specific evaluations are shown in Section 3.8 and 4.2.1 of Enclosure 5 to Reference 1, respectively. The evaluations concluded that operation of the HPCI and RCIC systems at the power uprate condition will not have any effect on the availability or the reliability of the systems. Power uprate operation will also not invalidate any of the pre-uprate design pressures or temperatures for the systems components. Compliance to GE SIL 480 is discussed in Section 4.2.1 for the HPCI system. Compliance to GE SIL 377 for the RCIC system is discussed previously in the response to item D.8.

The Control Rod Drive (CRD) hydraulic system plant-specific evaluation is shown in Section 2.5.1 of Enclosure 5 to Reference 1. The effectiveness of scram time performance during power uprate meets the ISTS requirements. The CRD pumps were evaluated against the 250 psi required minimum pressure differential between the drive water and the vessel bottom head and were found to have sufficient capacity.

The plant-specific evaluation of the RRS is shown in Section 3.4 of Enclosure 5 to Reference 1. The expected increase in the RRS operating pressure, temperature, drive motor horsepower, pump flow, pump speed and pump brake horsepower are within the current system design. There is no concern with vibration since there is no change to the total core flow and only about 1% increase in the recirculation pump speed (20 rpm increase). This RPM change has been evaluated and concluded to have no impact on the RRS vibration response for the power uprate condition. Please refer to response to item D.7 for additional discussion on RRS vibration.

The MSRV setpoints are conservatively analyzed at an increase of 35 psi at the power uprate condition to bound the proposed 30 psi increase in reactor dome pressure. Although the current BFN technical specification for MSRV opening setpoint tolerance is +1%, the power uprate analyses assumed an analytical value of +3% consistent with the proposed criteria in TS-386 (Reference 8). These MSRVs characteristics are used to demonstrate the plant-specific vessel overpressure protection capability (see Section 3.2 of Enclosure 5 to Reference 1). The power uprate performance evaluation for systems connected to the reactor coolant system boundary (HPCI, CS, RCIC and SLC) are also based on these MSRVs characteristics. Please see also responses to items D.1 and D.3.

The plant-specific analyses demonstrated that the fuel design criteria and fuel operating limits, such as

MAPLHGR, and OLMCPR will still be met at the power uprate level (Section 2 of Enclosure 5 to Reference 1). The reload licensing analyses will continue to meet acceptable NRC criteria as specified in GESTAR.

The plant-specific LOCA analyses were performed for a spectrum of recirculation line breaks using the NRC approved SAFER/GESTR methodology. The results are shown in Section 4.3 of Enclosure 5 to Reference 1 and confirmed the capability of the ECCS to maintain the fuel integrity within established licensing design criteria at the power uprate condition. The containment responses following postulated design basis accidents were also evaluated in Section 4.1.1 and 4.1.2 of Enclosure 5 to Reference 1. The results showed that the containment LOCA responses are within the existing containment design pressure and temperature limits. Radiological releases following postulated design basis accidents are evaluated on a plant-specific basis using methods consistent with Appendix H of NEDC-31897P-A and the Staff position paper. The results, shown in Section 9.2 of Enclosure 5 to Reference 1, are within the guidelines of 10 CFR 100.

The plant-specific limiting transients evaluations are shown in Section 9.1 of Enclosure 5 to Reference 1. Appendix E of NEDC-31897P-A is used to identify the limiting MCPR transient events for analyses. The procedures used are consistent with the NRC approved GESTAR methodology. Cycle specific reload licensing analyses are performed using the same procedures to confirm the cycle specific OLMCPR requirements.

The post-accident environmental evaluations for uprated conditions have not been completed. However, preliminary evaluations have been performed for equipment located inside the drywell. These preliminary evaluations indicate the drywell accident profile for uprated conditions is enveloped by the equipment test profiles. Due to conservatism in the current drywell accident profiles, preliminary calculations indicate there will be no reduction in margin in the accident degradation equivalency comparisons performed at uprated power conditions.

REFERENCES:

1. TVA letter to NRC dated October 1, 1997, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical

Specification (TS) Change TS-384 - Request For License Amendment for Power Uprate Operation

2. TVA letter to NRC dated March 16, 1998, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 Technical Specification (TS) No. 384 Supplement 1 - Request for License Amendment for Power Uprate Operation

3. TVA letter to NRC dated March 20, 1998, Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-384 - Request for License Amendment for Power Uprate Operation

4. TVA letter to NRC dated April 1, 1998, Browns Ferry Nuclear Plant (BFN) - Response to Request for Additional Information (RAI) Regarding Units 2 and 3 Technical Specification (TS) Change TS-384, - Request for License Amendment for Power Uprate Operation, (TAC Nos. M99711, M99712) and Resolution of Control Room Emergency Ventilation System (CREVS) Issues (TAC Nos. M83348, M83349, M83350)

5. TVA letter to NRC dated May 1, 1998, Browns Ferry Nuclear Plant (BFN) - Supplemental Response to Request for Additional Information (RAI) Regarding Units 2 and 3 Technical Specification (TS) Change TS - 384, - Request for License Amendment for Power Uprate Operation, (TAC Nos. M99711, M99712) and Resolution of Control Room Emergency Ventilation System (CREVS) Issues (TAC Nos. M83348, M83349, M83350)

6. NRC letter to TVA dated March 13, 1998, Browns Ferry Nuclear Plant, Units 2 and 3: Request for Additional Information Relating to Technical Specification Change No. TS-384 - Power Uprate Operation (TAC Nos. M99711 and M99712)

7. TVA letter to NRC dated July 10, 1996, Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Installation Schedule for the Long-Term Stability Solution For Generic Letter (GL) 94-02

8. TVA letter to NRC dated December 11, 1996, Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specification (TS) 386 - Proposed Change to Safety/Relief Valve (S/RV) Set Point Requirements for Reactor Coolant System Integrity, TS 2.2.A

ENCLOSURE 2
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 2 AND 3

BROWNS FERRY NUCLEAR PLANT (BFN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING UNITS 2 AND 3 TECHNICAL SPECIFICATION (TS) CHANGE TS - 384, REQUEST FOR LICENSE AMENDMENT FOR POWER UPRATE OPERATION (TAC NOS. M99711 AND M99712)

FSAR REVIEW MATRIX

X = Anticipated Change(s) Under Evaluation
N = No Change Anticipated

Chapter	Sections Affected																				
	.1	.2	.3	.4	.5	.6	.7	.8	.9	.10	.11	.12	.13	.14	.15	.16	.17	.18	.19	.20	.21
1	X	X	X	N	X	X	X	N	N	N	N	X									
2	N	N	N	X	X	N															
3	N	X	X	X	X	X	X	X													
4	N	X	X	X	X	X	X	X	X	N	X	N									
5	N	X	X																		
6	N	X	X	X	X	N															
7	N	X	X	X	X	N	X	X	X	X	X	X	N	N	N	X	X	N	N		
8	N	X	X	X	X	X	X	X	X	N											
9	N	X	N	N	X																
10	N	N	N	N	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
11	X	X	X	N	X	X	X	X	N												
12	N	X	X																		
13	N	N	N	N	N	N	N	N	N	X											
14	X	N	N	X	X	X	N	X	X												

Appendix	All Sections	Comments
A	N	Historical
B	X	Technical Specifications
C	X	
D	N	
E	N	
F	N	
G	N	Historical
H	N	Deleted
I	N	Historical
J	N	Historical
K	N	Historical
L	N	Historical
M	X	
N	X	Unit 3 Reload Analysis

ENCLOSURE 3
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 2 AND 3

BROWNS FERRY NUCLEAR PLANT (BFN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING UNITS 2 AND 3 TECHNICAL SPECIFICATION (TS) CHANGE TS - 384, REQUEST FOR LICENSE AMENDMENT FOR POWER UPRATE OPERATION (TAC NOS. M99711 AND M99712)

COMMITMENTS

1. TVA will provide the reply to Section C in a supplemental response.

2. Review and revision of the EOIs, which includes a review of all EOI variables and limit curves, for changes due to power uprate is an ongoing activity during the power uprate implementation phase. The review, update of the EOIs for any changes, and training are scheduled to be completed prior to the power uprate start-up for Browns Ferry Unit 3.