Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon.
Nuclear

10 CFR 50.90

RS-10-021

October 4, 2010

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject: License Amendment Request to Eliminate Main Steam Isolation Valve Closure and Low Condenser Vacuum Scram Functions during Startup Mode

References:

(1) General Electric Report NEDO-20697, "Bottled-Up Operation of a BWR," dated November 1974 (document was previously submitted by EGC in ADAMS Accession Number ML02361028)

(2) General Electric Nuclear Energy Service Information Letter 107, "Increasing Flexibility of Reactor Startups," dated October 31, 1974

(3) General Electric Nuclear Energy Report GE-NE-0000-0005-7308-01P, "Dresden Unit 2 and 3 – Elimination of MSIV Closure and Low Condenser Vacuum Scram Function During Startup Mode," Revision 2, dated December 2002 (non-Proprietary version of document was previously submitted by EGC in ADAMS Accession Number ML023610236)

(4) Letter from T. W. Simpkin (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment to Eliminate Main Steam Isolation Valve Closure and Low Condenser Vacuum Scram Functions During Startup Mode," dated December 20, 2002 (ADAMS Accession Numbers ML02361028 and ML023610236)

(5) Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for License Amendment to Eliminate Main Steam Isolation Valve Closure and Low Condenser Vacuum Scram Functions During Startup Mode," dated May 16, 2003 (ADAMS Accession Number ML031480564)

(6) Letter from Patrick R. Simpson (Exelon Generation Company, LLC to U.S. NRC, "Withdrawal of Request for License Amendment to Eliminate Main Steam Isolation Valve Closure and Low Condenser Vacuum Scram Functions During Startup Mode," dated October 1, 2003 (ADAMS Accession Number ML032810614)

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to renewed Facility Operating License (FOL) Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS), Units 2 and 3, respectively.

The proposed amendment revises the applicability of FOL Appendix A, Technical Specification (TS) 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 5 (i.e., "Main Steam Isolation Valve – Closure") and Function 10 (i.e., "Turbine Condenser Vacuum – Low") to enable implementation of a modification that will eliminate these functions while in Mode 2 with reactor pressure greater than or equal to 600 psig.

In addition, the proposed amendment deletes TS Table 3.3.1.1-1 footnote (c), and Required Action F.2 of TS 3.3.1.1, consistent with the revised applicability for Functions 5 and 10. Finally, the proposed amendment renames TS Table 3.3.1.1-1 footnote (d) to footnote (c). No changes are being made to the required number of channels per trip system, surveillance requirements, or allowable values for these functions during Mode 1 operation.

The DNPS TSs currently require a reactor scram if vessel pressure is greater than or equal to 600 psig with the reactor mode switch in startup and the main steam line isolation valves (MSIVs) closed, or with a main condenser vacuum low condition. This scram logic was installed at DNPS as a result of experience gained during the startup of an early vintage boiling water reactor (BWR) in 1966 when operators had difficulty controlling reactor power above approximately 600 psig without pressure control. The operating condition under which this is a concern (i.e., Mode 2 startup with MSIVs closed or pressure control otherwise unavailable) is referred to as "bottled-up" startup operations and is described in Reference 1.

Experience on plant startups with later BWR designs indicated that this early experience was not inherent to the BWR design and so, in order to demonstrate this, a test was conducted at the Browns Ferry plant in 1974 to assess the susceptibility of this later BWR design to this same pressure and power oscillations, which is also described in Reference 1.

The conclusions from the 1974 Browns Ferry test were (1) that the later BWR design could be controlled in a bottled-up condition at reactor vessel dome pressures well in excess of 600 psig and (2) that there was no reason for this to be an unacceptable operating region and require a reactor scram when vessel dome pressure exceeds 600 psig with the MSIVs closed.

Subsequently, after initial startup of DNPS, GE Nuclear concluded in Services Information Letter (SIL) No. 107, "Increasing Flexibility of Reactor Startups," (Reference 2) that the results of the Browns Ferry test could be extended to cover all BWR/4 plants, but that the test results could not be generically applied to BWR/1, 2, or 3 plants without a test or engineering analysis.

Based on the GE recommendation in Reference 2, EGC analyzed, with GE, the applicability of Reference 2 to DNPS Units 2 and 3, which are both BWR/3 plants. Reference 3 documented this engineering analysis and concluded that the requirement to establish pressure control prior to exceeding 600 psig reactor vessel dome pressure could be eliminated for the DNPS units. In summary, Reference 3 stated that the test of the later BWR (i.e., Browns Ferry Unit 1) is applicable to the DNPS conditions for startup mode, up to and including the maximum design pressure.

In Reference 4, EGC submitted a license amendment request (LAR) to eliminate the applicability requirement for TS 3.3.1.1 Functions 5 and 10 while in Mode 2 with the reactor pressure greater than or equal to 600 psig. The LAR was based on, and provided, the Reference 3 engineering analysis which demonstrated the Browns Ferry test result (i.e., as described in Reference 2) was applicable to DNPS.

During review of this LAR, the NRC transmitted a request for additional information concerning the LAR. In response, EGC provided a quantifiable comparison of relevant plant parameters from DNPS to the same plant parameters from Browns Ferry, as well as from the early dual-cycle BWR that first experienced the problem (Reference 5). However, both the NRC and EGC recognized that an additional engineering analysis of the Browns Ferry test report would be required to demonstrate the applicability of the Browns Ferry conclusions to DNPS Units 2 and 3. As a result, EGC withdrew the LAR in Reference 6.

Since withdrawal of the original LAR in 2003, EGC has developed, with GE Hitachi Nuclear Energy (GEH), and in a timeframe commensurate with the DNPS business planning process, the engineering analysis that is required to support the original proposed TS change.

Therefore, EGC is requesting NRC review and approval of an LAR to eliminate the requirement for a reactor scram if vessel pressure is greater than or equal to 600 psig, with the reactor mode switch in startup and the MSIVs closed or with a main turbine condenser vacuum low condition. This LAR is based on, and includes the most recent engineering analysis prepared by GEH.

Attachment 1 to this letter provides an evaluation supporting the proposed changes. The marked-up TS pages, with the proposed changes indicated, are provided in Attachment 2 to this letter. Attachment 3 provides the marked-up copy of the affected Bases pages, which are provided for informational purposes.

Attachment 4 to this letter provides GEH Report 0000-0090-6825-R0-P, "Dresden Units 2 and 3 TRACG Analysis to Support Elimination of Mode 2 Scram Requirement," dated May 2009, which GEH considers to contain proprietary information. The proprietary information is identified by bracketed and underlined text. GEH requests that the proprietary information in Attachment 4 be withheld from public disclosure, in accordance with the requirements of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4). A signed affidavit supporting this request is provided in Attachment 4 to this letter. Attachment 5 to this letter provides a non-proprietary version of the GEH Report (i.e., 0000-0090-6825-R0-NP).

There are no Regulatory Commitments associated with this proposed amendment.

The proposed change has been reviewed by the DNPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program. EGC requests approval of the proposed change by October 5, 2011, with the amendment being implemented during the first outage of sufficient duration following approval.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," EGC is notifying the State of Illinois of this application for a change to the TS by sending a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this letter, please contact Mr. Timothy Byam at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 4th day of October 2010.

Respectfully,

Jeffrey L. Hansen
Manager - Licensing
Exelon Generation Company, LLC

Attachments:

Attachment 1:	Evaluation of Proposed Changes
Attachment 2:	Mark-up of Proposed Technical Specification Pages
Attachment 3:	Mark-up of Proposed Technical Specification Bases Pages
Attachment 4	Dresden Units 2 and 3 TRACG Analysis to Support Elimination of Mode 2 Scram Requirement (Proprietary)
Attachment 5:	Dresden Units 2 and 3 TRACG Analysis to Support Elimination of Mode 2 Scram Requirement (Non-Proprietary)

ATTACHMENT 1
Evaluation of Proposed Changes

ATTACHMENT 1
Evaluation of Proposed Changes

1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to renewed Facility Operating License (FOL) Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS), Units 2 and 3, respectively. The proposed amendment revises the applicability of FOL Appendix A, Technical Specification (TS) 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 5, "Main Steam Isolation Valve – Closure," and Function 10, "Turbine Condenser Vacuum – Low," to eliminate the requirement for these functions to be operable while in Mode 2 with reactor pressure greater than or equal to 600 psig.

In addition, the proposed amendment deletes TS Table 3.3.1.1-1 footnote (c) and Required Action F.2 of TS 3.3.1.1, consistent with the revised applicability for Functions 5 and 10. Finally, the proposed amendment renames TS Table 3.3.1.1-1 footnote (d) to footnote (c). No changes are being made to the required number of channels per trip system, surveillance requirements, or allowable values for these functions during Mode 1 operation.

These proposed changes will reduce the risk of an unnecessary plant transient while in Mode 2 by eliminating the potential for a reactor scram due to a single point failure on one pressure switch or RPS bus.

2.0 PROPOSED CHANGES

2.1 TS Table 3.3.1.1-1, Function 5 and Function 10, Column 2, "Applicable Modes or Other Specified Conditions," will be revised to remove the term "$2^{(c)}$."

2.2 TS Table 3.3.1.1-1 will be revised to remove footnote (c) "With reactor pressure > 600 psig," on pages 3.3.1.1-9 and 3.3.1.1-10.

2.3 TS Table 3.3.1.1-1, Function 7.a, "Scram Discharge Level," Column 5, Surveillance Requirements," will be revised to change the footnote designation from (d) to (c).

2.4 TS Table 3.3.1.1-1 will be revised to rename footnote (d) "Specified SR performance only required for Unit 3," to (c).

2.5 TS 3.3.1.1, Required Action F will be revised to delete Required Action F.2.

3.0 TECHNICAL EVALUATION

TS Table 3.3.1.1-1, Functions 5 and 10 are required to be operable in Mode 1 and Mode 2 with reactor pressure greater than or equal to 600 psig. For these functions, Table 3.3.1.1-1 requires Condition F of TS 3.3.1.1 to be entered if the required channels are not restored to operable status, or placed in trip (or associated trip system placed in trip) within the allowed Completion Time. The required action for Condition F is to be in Mode 2 within 8 hours, and reduce reactor pressure to less than 600 psig within eight hours. The basis for these required actions is to place the plant in a Mode or other specified condition in which the limiting condition for operation does not apply.

TS Table 3.3.1.1-1, Function 5, "Main Steam Isolation Valve – Closure," is provided to initiate a reactor scram on an MSIV closure. MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on an MSIV closure before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. Function 5 is required to be operable in Mode 1 and Mode 2 with reactor pressure greater than or equal to 600 psig since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In Mode 2 and reactor pressure less than 600 psig, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection. This function is automatically bypassed with the reactor mode switch in any position other than run and reactor pressure less than 600 psig.

TS Table 3.3.1.1-1, Function 10, "Turbine Condenser Vacuum – Low," is provided to shut down the reactor and reduce the energy input to the main condenser to help prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. For the loss of condenser vacuum event, the reactor scram reduces the amount of energy required to be absorbed by the main condenser and helps to ensure the minimum critical power ratio safety limit is not exceeded by reducing the core energy prior to the fast closure of the turbine stop valves. This function helps maintain the main condenser as a heat sink during this event. Function 10 is required to be operable in Mode 1 and Mode 2 when reactor pressure is greater than or equal to 600 psig since, in these Modes, a significant amount of core energy can be rejected to the main condenser. During Mode 2 with reactor pressure less than 600 psig, and Modes 3, 4, and 5, the core energy is significantly lower. This function is automatically bypassed with the reactor mode switch in any position other than run and reactor pressure is less than 600 psig.

For these two functions, Table 3.3.1.1-1 requires entry into TS 3.3.1.1, Condition F if the required channels are not restored to operable status or placed in trip (or associated trip system placed in trip) within the allowed Completion Time. The required action for Condition F is to be in Mode 2 within eight hours, and reduce reactor pressure to less than 600 psig within eight hours. The basis for these required actions is to place the plant in a Mode or other specified condition in which the limiting condition for operation does not apply.

The scram logic described above (i.e., TS Table 3.3.1.1-1 Functions 5 and 10) is the result of experience gained during the startup of an early vintage BWR in 1966 when operators had difficulty controlling reactor power above approximately 600 psig without pressure control. The operating condition under which this is a concern (i.e., Mode 2 startup with MSIVs closed or pressure control otherwise unavailable) is referred to as "bottled-up" startup operations and is described in Reference 1. Experience on plant startups with later BWR designs indicated that this early experience was not inherent to the BWR design and so, in order to demonstrate this, a test was conducted at the Browns Ferry plant in 1974 to assess the susceptibility of this later BWR design to this same pressure and power oscillation.

The conclusions from the 1974 Browns Ferry test were (1) that the later BWR design could be controlled in a bottled-up condition at reactor vessel dome pressures well in excess of 600 psig and (2) that there was no reason for this to be an unacceptable operating region and require a reactor scram when vessel dome pressure exceeds 600 psig with the MSIVs closed.

Subsequently, after initial startup of DNPS, GE Nuclear concluded in Services Information Letter (SIL) No. 107, "Increasing Flexibility of Reactor Startups," (Reference 2) that the scram requirement be eliminated following the successful 1974 tests during startup of the Browns Ferry plant and that the results of the Browns Ferry test could be extended to cover all BWR/4 plants, but that the test results could not be generically applied to BWR/1, 2, or 3 plants without a test or engineering analysis. DNPS is a BWR/3 design.

As noted above, the GE test report for the Browns Ferry testing (Reference 1) and SIL-107 (Reference 2) both indicate that the elimination of the scram requirement for the MSIV closure and low condenser vacuum functions in Mode 2 with the reactor pressure greater than or equal to 600 psig can be directly applied to only the BWR/4 type plant. Reference 2 also stated that the BWR/4 plant types were regarded as the first standardized designs, whereas the earlier BWR/1, BWR/2 and BWR/3 plants were regarded as plants of unique design. This standardized plant concept made it difficult for GE to generically extend the test results from Browns Ferry to earlier BWRs, as well as potentially different future designs. Therefore, Reference 2 recommended that earlier BWR plants be considered on an individual basis to determine bottled-up operating capability by a similar test procedure or engineering analysis.

Based on the GE recommendation in Reference 2, EGC analyzed, with GE, the applicability of Reference 2 to DNPS Units 2 and 3, which are both BWR/3 plants. Reference 3 documented this analysis and concluded that the requirement to establish pressure control prior to exceeding 600 psig reactor vessel dome pressure could be eliminated for the DNPS units. In summary, Reference 3 stated that the test of the later BWR (i.e., Browns Ferry Unit 1) is applicable to the DNPS conditions for startup mode, up to and including the maximum design pressure.

In Reference 4, EGC submitted a license amendment request (LAR) to eliminate the applicability requirement for TS 3.3.1.1 Functions 5 and 10 while in Mode 2 with the reactor pressure greater than or equal to 600 psig. The LAR was based on, and provided the Reference 3 analysis that demonstrated the Browns Ferry test result (i.e., as described in References 1 and 2) was applicable to DNPS.

During review of this LAR, the NRC transmitted a request for additional information concerning the LAR. In response, EGC provided a quantifiable comparison of relevant plant parameters from DNPS to the same plant parameters from Browns Ferry, as well as from the early dual-cycle BWR that first experienced the problem (Reference 5). However, both the NRC and EGC recognized that additional technical analysis of the Browns Ferry test report would be required to demonstrate the applicability of the Browns Ferry conclusions to DNPS Units 2 and 3. As a result, EGC withdrew the LAR in Reference 6.

Since withdrawal of the original LAR in 2003, EGC has developed, with GE Hitachi Nuclear Energy (GEH), and in a timeframe commensurate with the DNPS business planning process, the analysis that is required to support the original proposed TS change. This new analysis is provided in Attachment 4 to this letter.

The new analysis utilized the NRC-approved TRACG code model of DNPS to evaluate the susceptibility of Units 2 and 3 to power/flow oscillations as reactivity changes are introduced at low power and moderate to high pressures. This analysis examined a range of operating conditions outside the range of the test conditions, but still within the range of concern to demonstrate that the DNPS plant design is not susceptible to the phenomena experienced by the dual-cycle early BWR design plant. Attachment 4 to this letter documents the results and conclusions of the analysis which was performed for the MSIV closure function since the transient associated with the MSIV closure function is bounding for the low condenser vacuum function.

As documented in Attachment 4, this analysis concludes that, because of the stability exhibited by the DNPS TRACG model under operating conditions of concern, the TS requirement to scram the reactor when in Mode 2 with the MSIVs closed or main condenser vacuum low and vessel dome pressure greater than or equal to 600 psig can be eliminated for DNPS without introducing any consequences that might be adverse to safe plant operation.

Specifically, the reactor scram associated with the MSIV closure and low condenser vacuum (i.e., Functions 5 and 10 of TS 3.3.1.1) is in anticipation of the loss of loss of the normal heat sink and subsequent overpressurization transient. The existing scram logic is the result of experience gained during startup of an early vintage boiling water reactor in 1966 when operators had difficulty controlling reactor power above approximately 600 psig without pressure control. Experience on later plant startups indicates that the early experience may not be inherent to later boiling water reactor designs. As such, GEH subsequently recommended elimination of the Mode 2 scram requirement.

In Mode 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection from an overpressurization transient. During normal power ascension in Mode 2 with the MSIVs open, reactor pressure vessel (RPV) pressure is controlled by the pressure regulator with increasing pressure setpoints. The maximum pressure regulator setpoint, which would translate to 1000 psig at rated power, would only allow a maximum dome pressure of approximately 900 psig in the Mode 2 power range. The potential scenario in Mode 2 whereby the MSIVs would close unexpectedly and cause the pressure to increase would lead to the Average Power Rate Monitors, Neutron Flux-High, Setdown scram (i.e., TS 3.3.1.1, Function 2.a), followed by the Reactor Vessel Steam Dome Pressure-High scram (i.e., TS 3.3.1.1, Function 3).

Upon approval of this amendment request, at the next outage of sufficient duration on both DNPS Unit 2 and Unit 3, EGC will install a modification that will remove the portion of the RPS logic which ensures that Functions 5 and 10 are operable while in Mode 2 with pressure greater than or equal to 600 psig. Functions 5 and 10 will continue to be required in Mode 1.

4.0 REGULATORY ANALYSIS

4.1 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS), Units 2 and 3, respectively. The proposed amendment revises the applicability of Appendix A, Technical Specification (TS) 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 5 (i.e., Main Steam Isolation Valve – Closure) and Function 10 (i.e., Turbine Condenser Vacuum – Low) to eliminate the requirement for these functions to be operable while in Mode 2 with reactor pressure greater than or equal to 600 psig. In addition, the proposed amendment also deletes Required Action F.2 of TS 3.3.1.1 to align with the revised applicability for Functions 5 and 10. No changes are being made to the required number of channels per trip system, surveillance requirements, or allowable values for these functions during Mode 1 operation.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No.

The proposed changes to the DNPS Units 2 and 3 TS revise the applicability of two protective functions and delete the associated TS Action statement. TS requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Specifically, the reactor scram associated with the MSIV closure and low condenser vacuum (i.e., Functions 5 and 10 of TS 3.3.1.1) is in anticipation of the loss of the normal heat sink and subsequent overpressurization transient. The scram at high pressure in startup conditions when MSIVs close and/or main condenser vacuum is low does not impact the limiting accident or transient analyses. An analysis by General Electric Hitachi Nuclear Energy (GEH) demonstrated that the Mode 2 scram function for MSIV closure and low condenser vacuum can be eliminated without affecting safe plant operation. Elimination of these required scrams will not involve an increase in the probability of an accident previously evaluated.

Additionally, these proposed changes will not increase the consequences of an accident previously evaluated because the proposed changes do not adversely impact structures, systems, or components. These changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients by the plant safety analysis.

Function 5 is currently required in Mode 2 with reactor pressure greater than or equal to 600 psig to ensure that the reactor is shut down, thus helping to prevent an overpressurization transient due to closure of main steam isolation valves. Similarly, Function 10 is currently required in Mode 2 with reactor pressure greater than or equal to 600 psig to help prevent an overpressurization transient by anticipating the turbine stop valve closure scram on loss of condenser vacuum.

The existing scram logic is the result of experience gained during startup of an early vintage boiling water reactor in 1966 when operators had difficulty controlling reactor power above approximately 600 psig without pressure control. Experience on later plant startups indicates that the early experience may not be inherent to later boiling water reactor designs. As such, GEH subsequently recommended elimination of the Mode 2 scram requirement.

In Mode 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection from an overpressurization transient. During normal power ascension in Mode 2 with the MSIVs open, reactor pressure vessel (RPV) pressure is controlled by the pressure regulator with increasing pressure setpoints. The maximum pressure regulator setpoint, which would translate to 1000 psig at rated power, would only allow a maximum dome pressure of approximately 900 psig in the Mode 2 power range. The potential scenario in Mode 2 whereby the MSIVs would close unexpectedly and cause the pressure to increase would lead to the Average Power Rate Monitors, Neutron Flux-High, Setdown scram (i.e., TS 3.3.1.1, Function 2.a), followed by the Reactor Vessel Steam Dome Pressure-High scram (i.e., TS 3.3.1.1, Function 3).

The consequences of a previously analyzed event are dependent on the initial conditions assumed in the analysis, the availability and successful functioning of equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The consequences of a previously evaluated accident are not significantly increased by the proposed change. The proposed change does not affect the performance of any equipment credited to mitigate the radiological consequences of an accident. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed changes to the DNPS Units 2 and 3 TS revise the applicability of two protective functions and delete the associated TS Action statement. The RPS functions are not an initiator of any accident. Rather, the RPS is designed to initiate a reactor scram when one or more monitored parameters exceed their specified limits to preserve the integrity of the fuel cladding and the reactor coolant pressure boundary and minimize the energy that must be absorbed following an accident. The proposed changes do not alter the applicability for RPS functions during plant conditions in which an overpressurization transient is assumed to occur. Specifically, no changes are being made to the required number of channels per trip system, surveillance requirements, or allowable values for these functions during Mode 1 operation.

   The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not change or introduce any new equipment, modes of system operation or failure mechanisms.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No.

   Margins of safety are established in the design of components, the configuration of components to meet certain performance parameters, and in the establishment of setpoints to initiate alarms and actions. The proposed changes revise the applicability for Functions 5 and 10 of TS 3.3.1.1 and delete an associated TS Action Statement. The proposed changes do not alter the applicability for RPS functions during plant conditions in which an overpressurization transient is assumed to occur.

In addition, the proposed changes do not affect the probability of failure or availability of the affected instrumentation. Furthermore, the proposed changes will reduce the probability of test-induced plant transients and equipment failures.

The proposed changes to the applicability for Functions 5 and 10 of TS 3.3.1.1 have no impact on equipment design or fundamental operation. There are no changes being made to safety limits or safety system allowable values that would adversely affect plant safety. The performance of the systems important to safety is not significantly affected by the proposed changes. The proposed change does not affect safety analysis assumptions or initial conditions and therefore, the margin of safety in the original safety analyses is maintained.

As documented above, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above discussion, EGC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4.2 Applicable Regulatory Requirements/Criteria

EGC has evaluated the proposed changes to Appendix A, Technical Specifications (TS) of Renewed Facility Operating License Nos. DPR-19 and DPR-25 to determine compliance with applicable regulations and regulatory requirements. EGC has determined that the proposed changes to revise the applicability of TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 5, "Main Steam Isolation Valve - Closure" and Function 10, "Turbine Condenser Vacuum - Low"), and the proposed change to delete Required Action F.2 of TS 3.3.1 do not require any exemptions or relief from regulatory requirements, other than a revision to the TS. EGC has also determined that the proposed changes do not affect conformance with any General Design Criteria differently than described in the DNPS Updated Final Safety Analysis Report. The regulatory bases and guidance documents associated with the systems discussed in this license amendment request are described below:

10 CFR 50.36, "Technical Specifications"

10 CFR 50.36, "Technical specifications," requires that the facility's TS will include a section addressing limiting conditions for operation (LCO). In accordance with 10 CFR 50.36(d)(2)(ii), the limiting condition for operation of a nuclear reactor must be established for each item meeting one or more the specified criteria. One of these criteria is Criterion 3 which requires an LCO for a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The RPS is designed to initiate a reactor scram when one or more monitored parameters exceed their specified limits to preserve the integrity of the fuel cladding and the reactor coolant pressure boundary and minimize the energy that must be absorbed following an accident. While the proposed change eliminates the requirement for a scram in Mode 2 when the MSIVs are closed or the condenser vacuum is low with vessel pressure greater than or equal to 600 psig, the proposed changes do not alter the applicability for RPS functions during plant conditions in which an overpressurization transient is assumed to occur. Specifically, no changes are being made to the required number of channels per trip system, surveillance requirements, or allowable values for these functions during Mode 1 operation DNPS will continue to meet the requirements of 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

DNPS Units 2 and 3 were originally designed and constructed prior to the issuance of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," (GDC). Proposed GDC were issued in July 1967, during the construction of the plants. These proposed criteria were not yet adopted as regulatory requirements at the time DNPS was built and licensed. Nevertheless, the proposed GDC were used by the Atomic Energy Commission to evaluate the original design of DNPS Units 2 and 3. This engineering analysis indicated that, based on the applicant's understanding of the intent of the proposed GDC, DNPS fully satisfies the intent of the criteria. The DNPS Updated Final Safety Analysis Report (UFSAR) Section 3.1.1 addresses DNPS's conformance to the proposed GDC that were issued in July 1967. DNPS UFSAR Section 3.1.2 provides the results of a later evaluation of DNPS Unit 2 evaluated against the final GDC that were published in July 1971.

10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The proposed changes have been evaluated against General Design Criteria 13, 20, 21, 22, 23, 24, 25, 26, and 29. EGC has verified that these changes do not affect the ability of DNPS to continue monitoring variables and systems for normal operation, anticipated operational occurrences (AOOs), and accident conditions, as well as controlling reactivity. The proposed changes simply eliminate the requirement to scram the reactor for Functions 5 and 10 of TS Table 3.3.1.1-1 while the reactor is in Mode 2. The proposed changes do not affect these functions in Mode 1 when an overpressurization transient is assumed to occur. Therefore, DNPS will continue to be in compliance with these General Design Criteria.

4.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

EGC has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. General Electric Report NEDO-20697, "Bottled-Up Operation of a BWR," dated November 1974 (document was previously submitted by EGC in ADAMS Accession Number ML02361028)
2. General Electric Nuclear Energy Services Information Letter No. 107, "Increasing Flexibility of Reactor Startups," dated October 31, 1974
3. General Electric Nuclear Energy Report GE-NE-0000-0005-7308-01P, "Dresden Unit 2 and 3 – Elimination of MSIV Closure and Low Condenser Vacuum Scram Function During Startup Mode," Revision 2, dated December 2002 (non-Proprietary version of document was previously submitted by EGC in ADAMS Accession Number ML023610236)
4. Letter from T. W. Simpkin (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment to Eliminate Main Steam Isolation Valve Closure and Low Condenser Vacuum Scram Functions During Startup Mode," dated December 20, 2002 (ADAMS Accession Numbers ML02361028 and ML023610236)
5. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for License Amendment to Eliminate Main Steam Isolation Valve Closure and Low Condenser Vacuum Scram Functions During Startup Mode," dated May 16, 2003 (ADAMS Accession Number ML031480564)
6. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Withdrawal of Request for License Amendment to Eliminate Main Steam Isolation Valve Closure and Low Condenser Vacuum Scram Functions During Startup Mode," dated October 1, 2003 (ADAMS Accession Number ML032810614)

ATTACHMENT 2

Mark-up of Proposed Technical Specification Pages

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25

<u>REVISED TECHNICAL SPECIFICATION PAGES</u>

3.3.1.1-2
3.3.1.1-9
3.3.1.1-10

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 38.5% RTP	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	8 hours
	~~AND~~		
	~~F.2~~	~~NOTE~~ ~~Only required to be met for Function 5, Main Steam Isolation Valve Closure, and Function 10, Turbine Condenser Vacuum Low.~~	
		~~Reduce reactor pressure to < 600 psig.~~	~~8 hours~~

(continued)

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux-High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 122% RTP
d. Inop	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.18	NA
3. Reactor Vessel Steam Dome Pressure-High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1045 psig
4. Reactor Vessel Water Level-Low	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 2.65 inches
5. Main Steam Isolation Valve-Closure	1~~, 2(c)~~	8	F	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed
6. Drywell Pressure-High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1.94 psig

(continued)

~~(c) With reactor pressure ≥ 600 psig.~~

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level–High					
a. Thermal Switch (Unit 2) Level Indicating Switch (Unit 3)	1,2	2	G	SR 3.3.1.1.1~~(d)~~ SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12~~(d)~~ SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 38.7 gallons (Unit 3)
	5(a)	2	H	SR 3.3.1.1.1~~(d)~~ SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12~~(d)~~ SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 38.7 gallons (Unit 3)
b. Differential Pressure Switch	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 38.7 gallons (Unit 3)
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 38.7 gallons (Unit 3)
8. Turbine Stop Valve-Closure	≥ 38.5% RTP	4	E	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure–Low	≥ 38.5% RTP	2	E	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 466 psig
10. Turbine Condenser Vacuum–Low	1~~, 2(c)~~	2	F	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 20.5 inches Hg vacuum
11. Reactor Mode Switch–Shutdown Position	1,2	1	G	SR 3.3.1.1.16 SR 3.3.1.1.18	NA
	5(a)	1	H	SR 3.3.1.1.16 SR 3.3.1.1.18	NA
12. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.18	NA
	5(a)	1	H	SR 3.3.1.1.8 SR 3.3.1.1.18	NA

(c)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
~~(c) With reactor pressure ≥ 600 psig.~~
~~(d)~~ Specified SR performance only required for Unit 3.

(c)

ATTACHMENT 3

Mark-up of Proposed Technical Specification Bases Pages

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25

REVISED BASES PAGES
<u>(PROVIDED FOR INFORMATION ONLY)</u>

B 3.3.1.1-16
B 3.3.1.1-20

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Main Steam Isolation Valve—Closure (continued)

valid signal. This Function is only required in MODE 1 ~~and MODE 2 with reactor pressure greater than or equal to 600 psig~~ since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In MODE 2 ~~and reactor pressure less than 600 psig~~, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection. This Function is automatically bypassed with the reactor mode switch in any position other than run ~~and reactor pressure is less than 600 psig~~.

6. Drywell Pressure—High

High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. The Drywell Pressure—High Function is assumed to scram the reactor for LOCAs inside the primary containment.

The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure switches that sense drywell pressure. The Allowable Value was selected to be as low as possible and indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure—High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required in MODES 1 and 2 where considerable energy exists in the RCS, resulting in the limiting transients and accidents.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

10. Turbine Condenser Vacuum—Low

The Turbine Condenser Vacuum—Low Function is provided to shut down the reactor and reduce the energy input to the main condenser to help prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. Loss of condenser vacuum occurs when the condenser can no longer handle the heat input (e.g., loss of heat transfer capability or excessive inleakage). This condition initiates a closure of the turbine stop valves and turbine bypass valves, which eliminates the reactor heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise and an increase in fuel surface heat flux. To prevent the fuel cladding integrity Safety Limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the fuel cladding integrity Safety Limit from being exceeded, in the event of a turbine trip transient with bypass closure. The condenser low vacuum scram is anticipatory to the turbine stop valve closure scram.

Turbine condenser vacuum pressure signals are derived from four pressure switches that sense the pressure in the condenser. The Allowable Value is consistent with the main turbine trip on the low main condenser vacuum setpoint, and provides main condenser overpressure protection by shutting down the reactor; thereby, reducing energy into the main condenser.

Four channels of Turbine Condenser Vacuum—Low Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is only required in MODE 1 ~~and MODE 2 when reactor pressure is ≥ 600 psig since~~, since in this MODE, there is ~~in these MODES,~~ a significant amount of core energy that can be rejected to the main condenser. During ~~MODE 2 with reactor pressure < 600 psig, and~~ MODES 2, 3, 4, and 5, the core energy is significantly lower. This Function is automatically bypassed with the reactor mode switch in any position other than run ~~and reactor pressure is < 600 psig~~.

(continued)

ATTACHMENT 5

GEH Report 0000-0090-6825-R0-NP
Dresden Units 2 and 3 TRACG Analysis to Support Elimination of
Mode 2 Scram Requirement (Non-Proprietary)

HITACHI

GE Hitachi Nuclear Energy

0000-0090-6825-R0-NP
DRF 0000-0064-9061
Revision 0
Class I
May 2009

Non-Proprietary Information

Licensing Report

Dresden Units 2 and 3
TRACG Analysis to Support Elimination of Mode 2 Scram Requirement

Principal Contributors:
D. W. Hiltbrand
J. L. Casillas
R. H. Jacobs
C. L. Heck

INFORMATION NOTICE

This document is a non-proprietary version of 0000-0090-6825-R0-P, which has the proprietary information removed. Portions of the document that have been removed are indicated by double open and closed brackets as shown here [[ ]].

IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

Please Read Carefully

REVISION SUMMARY

Rev	Required Changes to Achieve Revision
0	NA

TABLE OF CONTENTS

1.0 INTRODUCTION and BACKGROUND

1.1 Objective

This report presents a study to demonstrate the acceptable performance of TRACG for simulating low power phenomena in support of Dresden's Units 2 and 3 license amendment to eliminate the Technical Specification scram requirement when in Mode 2 with Main Steam Isolation Valves (MSIVs) closed and vessel dome pressure greater than 600 psig. The objective of the study is to demonstrate that over the operating conditions range of interest (1) the TRACG code provides a reasonable and realistic simulation of the Dresden plant, and (2) the TRACG model of the Dresden plant exhibits no oscillatory tendencies when subjected to significant reactivity and vessel dome pressure perturbations. The first objective was accomplished by comparing the Dresden TRACG model response to plant data for a test conducted at Browns Ferry in 1974 that was used to support eliminating this same scram requirement at BWR/4 plants. A successful outcome for this objective also demonstrated the applicability of the Browns Ferry test results to the Dresden plant. With the physical reasonableness of the model established, the Dresden TRACG model was then subjected to additional reactivity and vessel dome pressure perturbations to further demonstrate the plant's stability via the model's response. Use of this report is limited to startup operation only with reactor thermal power less than 5% of rated and the vessel dome pressure greater than 600 psig, but less than 1005 psig. This power level is less than that required for thermal limits monitoring, and therefore, thermal limits are not affected. [[

]]

1.2 Background

The Dresden Technical Specifications currently require a reactor scram if vessel dome pressure exceeds 600 psig with the reactor mode switch in STARTUP and the MSIVs closed. This requirement dates back to the startup of an earlier GEH dual-cycle BWR in 1966 when operators found it difficult to control power above 600 psig without pressure control. Experience on plant startups with later BWR designs indicated that this early experience was not inherent to the BWR design and so, in order to demonstrate this, a test was conducted at the Browns Ferry plant in 1974 to assess the susceptibility of this later BWR design to this same pressure and power oscillation. The operating condition under which this is a concern is referred to as "bottled-up" startup operations, i.e., Mode 2 startup with MSIVs closed or pressure control otherwise unavailable (Reference 1). The conclusions from the Browns Ferry test were (1) that the later BWR design could be controlled in a bottled-up condition at reactor vessel dome pressures well in excess of 600 psig and (2) that there was no reason for this to be an unacceptable operating region and require a reactor scram when vessel dome pressure exceeds 600 psig with the MSIVs closed. It was further concluded that the results of the Browns Ferry test could be extended to cover all BWR/4 plants, but that the test results would not be generically applied to BWR/1, 2, or 3 plants without a test or engineering analysis (Reference 2).

Given the results of the Browns Ferry test and to provide additional operating flexibility, GEH recommended that the bottled-up startup operation should be considered as an option to (but not as a replacement for) the more conventional pressure and steam flow heat-up with MSIVs open (Reference 2) and under Pressure Control mode. Consistent with the conclusions of Reference 1, Reference 2 also concluded that any older plant such as BWR/1, 2, or 3 wishing to pursue this option would require additional engineering analysis or testing.

Based on the Reference 2 recommendation, Exelon requested that GEH review the requirements for Dresden Units 2 and 3 (both BWR/3 plants) to maintain such a scram function. Reference 3 documents GEH's review and concludes that the requirement to establish pressure control prior to exceeding 600 psig reactor vessel dome pressure could be eliminated for the Dresden units. Exelon subsequently applied for a Technical Specification change for Dresden Units 2 and 3 to increase the setpoint vessel dome pressure in the STARTUP mode for the scram on MSIV isolation. In response to Exelon's Technical Specification change request, the NRC stated that additional, quantifiable information was required to demonstrate that Dresden would not be subject to the phenomena of concern. Reference 4 provided quantifiable information in the form of comparing plant parameters relevant to the phenomena between the Dresden units and Browns Ferry and between the Dresden units and the early dual-cycle BWR that experienced the problem behavior. The NRC responded that based on the conclusions in the Browns Ferry test report a detailed analysis was needed to support applicability of the Browns Ferry test to Dresden Units 2 and 3. The objective of the analysis described herein is to provide the detailed analysis required to demonstrate the applicability of the Browns Ferry test to the Dresden Units 2 and 3. In addition, the analysis examined a range of operating conditions outside the range of the test conditions, but still within the range of concern to demonstrate that the Dresden plant design is not susceptible to the phenomena experienced by the dual-cycle early BWR design plant.

2.0 REFERENCES

Item	Reference
1	NEDO-20697, "Bottled-Up Operation of a BWR", November 1974
2	SIL-107, "Increasing Flexibility of Reactor Startups S107", October 31, 1974
3	GE-NE-0000-0005-7308-01P Revision 2, "Dresden Unit 2 and 3 – Elimination of MSIV and Low Condenser Vacuum Scram During Startup Mode", December 2002.
4	GE-NE-0000-0014-1511 Revision 2, "Dresden Elimination of Low Pressure Isolation Setpoint – NRC RAI", May 15, 2003
5	GE-NE-0000-0030-9614-R1-P Revision 1, "Dresden Unit 3 TRACG Modeling for Transient HPCI Steam Line Flooding Evaluations", August 2005
6	Report 0000-0008-8536 ITEC-001054 Revision 2, "Zero Power Reactivity and Rod Worth Report for Dresden 3 Cycle 18", October 2002
7	Not Used.
8	NEDE-32176P Revision 4, "Licensing Topical Report TRACG Model Description", January 2008.
9	NEDE-32177P Revision 3, "TRACG Qualification", August 2007
10	TODI 08-006, Design Input for the Analysis to Removal of Scram in Mode 2 above 600 psig Reactor Pressure, 2/28/08.

3.0 SUMMARY and CONCLUSIONS

The analysis described herein uses a TRACG model of the Dresden plant to evaluate the susceptibility of Dresden Units 2 and 3 to power/flow oscillations as reactivity changes are introduced at low power and moderate to high pressures. A TRACG model of the Dresden plant that has been successfully benchmarked to actual plant transients is used as the starting model for this analysis. This tested TRACG model is then maneuvered from full power to low power and lower pressure conditions through a combination of recirculation pump speed reduction and rod insertion steps. The evaluation is then performed at these low power conditions in four distinct phases.

In the first phase, the Dresden TRACG model is benchmarked to the Browns Ferry test data by subjecting the model to a pressure perturbation caused by opening turbine bypass valves. The results of this phase demonstrated (1) that the TRACG model is suitable for simulating the low power conditions and phenomena of interest and (2) that the Dresden plant behavior is consistent with the Browns Ferry plant behavior when subjected to similar test conditions. In the second phase, the Dresden TRACG model pressure is decreased at this low power in order to evaluate the model's response at the lower end of the pressure range of interest. This phase demonstrated that, at the lower pressures, the model does not exhibit any power/flow oscillations as pressure-induced reactivity perturbations are applied. The third phase evaluated the Dresden plant response to reactivity perturbations introduced by single and multiple rod movements. This phase first demonstrated that the model's response to a single rod reactivity perturbation is comparable to the Browns Ferry plant response to a similar single rod reactivity perturbation test. This phase secondly confirmed the stable response when the model was then subjected to greater reactivity insertions by the rapid and slow movement of multiple rods. In all cases, the model response was well-behaved and the imposed perturbations were quickly damped. The results of these first three phases established that for pressure-induced and rod-induced reactivity perturbations at low power and moderate-to-high pressures, the model, and hence the Dresden plant, can be expected to respond in a well-behaved manner. The final phase entailed simulating the "bottled-up" state of the plant by closing the MSIVs. Under these conditions, as the closure of the MSIVs resulted in a rapid pressurization and its associated positive reactivity addition, the model remained well-behaved and quickly stabilized the power and flow.

The overall conclusion of this demonstration is that because of the well-behaved nature of the Dresden TRACG model when subjected to small and significant reactivity perturbations, the Dresden plant can be expected to respond in a similar manner such that continued safe control of the plant can be maintained in these Mode 2 conditions.

With the latter conclusion in hand, the Dresden TRACG model was analyzed further in order to cover the full range of operating conditions applicable to the current Technical Specification requirement and to provide additional assurances that the plant would remain stable when subjected to various transients in this Mode 2 configuration. These additional analyses were performed at several initial vessel dome pressures and included rapid depressurization and pressurization events as well as rapid and slow rod movement events. In all cases analyzed, the objective was simply to determine if the system exhibited any signs of power oscillations that might require further analysis or otherwise raise questions regarding the acceptability of eliminating this Technical Specification Mode 2 scram requirement at relatively low power levels up to approximately 5% of rated reactor thermal power. The results from all of the transients analyzed indicate that the plant can be expected to respond in a well-behaved fashion to significant system perturbations. In other words, the response of the TRACG model to several extreme perturbations did not provide any evidence of power oscillations that might lead to a phenomenon similar to the one experienced by the early design dual-cycle BWR.

The overall conclusion of this analysis is that, because of the stability exhibited by the Dresden TRACG model under the operating conditions of concern, the Technical Specification requirement to scram the reactor when in Mode 2 with MSIVs closed and vessel dome pressure greater than 600 psig can be eliminated for the Dresden plant without introducing any consequences that might be adverse to safe plant operation. Use of this report is limited to startup operation only with reactor thermal power less than 5% of rated and the vessel dome pressure greater than 600 psig, but less than 1005 psig. [[

]]

4.0 METHODOLOGY

The discussion below summarizes the TRACG computer code used for the calculations. The TRACG04A code version is used in this analysis. The application of this code is within the accepted code application capabilities and is described in Reference 8.

TRACG has been qualified for simulation of flow instability, hydraulic flow oscillations, and reactivity insertion phenomena in Reference 9. Although the application of TRACG herein is not considered a flow stability analysis, the citing of the flow oscillations and reactivity insertion phenomena qualification is intended to show the breadth of application of TRACG and provide some assurance that this application lies within the qualified scope of TRACG.

4.1 Inputs and Assumptions

TRACG Model Input:

The Dresden-3 Cycle 18 TRACG basedeck from Reference 5 was used as the starting point for developing the TRACG model for this analysis. This TRACG Dresden model was used in Reference 5 to perform benchmarks to two turbine trip events that occurred at Dresden in January 2004 and was concluded to provide a reasonable simulation of the plant behavior. Thus, there was already demonstrated evidence that the TRACG model to be used in this analysis provided a realistic simulation of the Dresden plant.

The channel grouping established in Reference 5 was used in this analysis. [[

]]

Therefore, the channel grouping from the original analysis was considered adequate to capture the neutronics feedback important to this analysis.

Plant Configuration Changes:

The plant configuration used for this calculation is applicable to the current plant configuration given in Reference 10, though the steam dryer and pressure control system changes are not included. This calculation is independent of steam dryer modification due to the very low steaming rate and pressure drop associated with these conditions. The calculation is also independent of the installation of digital EHC because pressure control is not credited in these calculations. [[

]]

Assumption 1:

The Browns Ferry test was conducted during a plant startup in which the reactor coolant is being heated up as power is increased; thus, the reactor power is primarily being used for the sensible heat addition to the coolant (and vessel/piping metal) as opposed to steam production. Steam production is assumed insignificant because, as identified on page A-3 of the test report (Reference 1), the turbines are isolated from the vessel and gland seal steam and the steam jet air ejectors (SJAE) are being supplied from the auxiliary boiler. Furthermore, since the reactor coolant is heating up, it is expanding and level control is accomplished by inventory letdown from the reactor water cleanup system. Any coolant addition is a result of leakage from the control rod drive system. The amount of leakage is small such that even if the additional water is introduced at a very low temperature, the amount of core inlet subcooling is therefore small as well. This assumption is supported by statements in the test documentation. Specifically, in the first paragraph on page 2 of the test report (Reference 1), reference is made to the "near-saturated condition of the vessel water at the operating point" and on page A-3 of the test report (Reference 1), Section 5 identifies the initial conditions stating that the control rod drive and cleanup systems are operating.

For this analysis, to achieve the desired low power initial condition, the Dresden TRACG model from a previous transient test benchmark at nearly full power was maneuvered to the low power Mode 2 condition. As a result, the coolant was already heated up and the reactor power was being used for latent heat addition to the coolant, thereby, generating steam. Since the reactor power was low (~ 1 to 2%), the amount of steam production was small, but nonetheless, coolant makeup was required in order to maintain level. To provide this makeup water, the feedwater fill junction was used in the model with the fill enthalpy controlled to maintain nearly saturated conditions at the core inlet. This fill path effectively simulated the injection of makeup water from the Reactor Water Cleanup System that would be introduced into the feedwater line in the real plant line-up test conditions. The larger magnitude of steam production compared to the test also required that a path exist for the steam to be discharged from the model. The discharge path used in the model was to maintain a small fixed demand on the turbine bypass valves. This required that the MSIVs remain open in this analysis, which was consistent with the test configuration for the pressure perturbation test, but not for the reactivity perturbation test. This difference was addressed in the analysis.

These differences in initial conditions between the Browns Ferry test and the Dresden TRACG model are judged to not have a significant impact on the comparison effort. [[

]]

Assumption 2:

Control rods have been grouped within the TRACG model using a Dresden 3 Cycle 18 control rod map that reflects the actual core design conditions given in Reference 6 for initial criticality as a starting guideline. See Figure 4.1 for a diagram of the control rod grouping applied in this TRACG model. The choice of ten control rod groups in the model provides the needed flexibility to achieve the desired initial conditions, namely just critical in Mode 2 at very low power and nearly saturated core inlet conditions. Control rod group 10 corresponds to a single control rod at position 34-27 for the control rod used for the reactivity perturbation. This control rod is in a similar central location in the core as that used in the Browns Ferry test for the reactivity perturbation. The magnitude of reactivity worth of the test control rod is considered in the analysis by performing a sensitivity study.

Figure 4.1: TRACG Control Rod Grouping

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R
15						3	5	4	5	3					
14				5	1	9	2	6	2	9	1	5			
13			5	3	5	4	5	3	5	4	5	3	5		
12		5	1	8	2	7	1	5	1	7	2	8	1	5	
11		3	5	4	5	3	5	4	5	3	5	4	5	3	
10	1	9	2	7	1	5	2	6	2	5	1	7	2	9	1
9	5	4	5	3	5	4	5	3	5	4	5	3	5	4	5
8	2	6	1	5	2	6	1	5	1	6	2	5	1	6	2
7	5	4	5	3	5	4	5	3	10	4	5	3	5	4	5
6	1	9	2	7	1	5	2	6	2	5	1	7	2	9	1
5		3	5	4	5	3	5	4	5	3	5	4	5	3	
4		5	1	8	2	7	1	5	1	7	2	8	1	5	
3			5	3	5	4	5	3	5	4	5	3	5		
2				5	1	9	2	6	2	9	1	5			
1						3	5	4	5	3					

— Control rods which are partially inserted in PANAC11 wrapup, fully withdrawn in Reference 6, and get fully inserted in TRACG model.

5 — Control rods which are fully inserted in PANAC11 wrapup, fully withdrawn in Reference 6, and untouched in TRACG model.

4 — Control rods which are fully withdrawn in PANAC11 wrapup, partially inserted in Reference 6, and get fully inserted in TRACG model.

— Control rods which are fully withdrawn in PANAC11 wrapup, fully withdrawn in Reference 6, and untouched in TRACG model.

3 — Control rods are fully withdrawn in PANAC11 wrapup, fully inserted in Reference 6, and get partially inserted in TRACG model.

10 — Control rod which is fully withdrawn in PANAC11 wrapup, fully withdrawn in Reference 6, and used for reactivity perturbation in TRACG model.

4.2 Evaluation

4.2.1 Evaluation Approach

In order to demonstrate the applicability of the Browns Ferry test results to the Dresden units, the TRACG model of Dresden Unit 3 from Reference 5 was maneuvered to operating conditions consistent with the test conditions for Browns Ferry. The TRACG model was then subjected to a pressure perturbation similar to that applied during the test and the response of the model compared to the test results. This computer run formed the "base case" for the analysis. The base case analysis results are presented in Section 5.2.1

Additional analyses were performed with the TRACG model to ensure that the scenario of concern was thoroughly evaluated. The additional analyses consisted of the following.

1. The TRACG model was initialized similar to the base analysis with the exception that the reactor pressure was established at pressures down to approximately 600 psia instead of 920 psia used in the test. This was necessary in order to evaluate the full pressure range of interest, especially considering that, since this event is sensitive to void reactivity, the largest phasic density differences occur at lower pressures. The same pressure perturbation as the Browns Ferry test was applied at this lower initial pressure condition and the response evaluated. This case is referred to as the "low pressure analysis". These results are presented in Section 5.2.2.

2. Using the lower pressure initial conditions that exhibited the larger response, the pressure perturbation was applied differently than was done in the test and in the base case. In the test and the base cases, the perturbation was applied such that negative reactivity was inserted first followed by a positive reactivity insertion. In this run, the perturbation was applied such that positive reactivity was inserted first, i.e., the pressure was increased first then decreased. This case is referred to as the "pressure-induced reactivity

analysis" and reflects a sudden reactivity insertion during a typical startup pressure increase scenario. These results are presented in Section 5.2.3.

3. Using the lower pressure initial conditions, a reactivity perturbation was applied by inserting and then withdrawing a single rod near the center of the core. The rod was inserted over one second and then withdrawn 15 seconds later over a 1 second interval. Following this run, a group of rods was rapidly withdrawn and then inserted to assess the stability of the model to a large reactivity swing. A third set of cases was then run starting from the end of these results. In this third set, one rod group was continuously and slowly withdrawn over the full transient time interval. The single rod run reflects a similar perturbation as performed in the Browns Ferry test. These runs are collectively referred to as the "rod-induced reactivity analysis". These results are presented in Section 5.2.4.

4. The final cases analyzed were started from the end of the second rod-induced reactivity cases (due to the very low power existing at the end of these runs). In these runs, the MSIVs were closed instantaneously at the start of the run. No rod groups were moved. The transient was used to evaluate the response of the model to the pressurization that occurs from the steam production in a 'bottled-up' condition. These cases are referred to as the "steam volume analysis" and reflect a scenario of continuous pressure increase during startup conditions. This scenario is the most relevant for the bottled up reactor with excess power and steaming rate. These results are presented in Section 5.2.5.

4.2.2 Evaluation Objectives

The Technical Specification requirement for establishing pressure control in a BWR beyond a reactor pressure of 600 psig protects against potential oscillations of pressure and power in Startup Mode 2. This pressure control requirement has been eliminated for standard BWR/4 and later designs based on the tests conducted in a large BWR/4 and documented in Reference 1. The TRACG simulations outlined above and documented in Section 5.2 are intended to demonstrate first that the Dresden plant would behave substantially similar to the key tests (Sections 5.2.1 and 5.2.4) In Reference 1, and second that the reactor under even larger pressure and power perturbations (Sections 5.2.2 and 5.2.3) does not exhibit any tendency to pressure and power oscillations.

In all cases analyzed, the objective was simply to determine if the system exhibited any signs of power oscillations that might require further analysis or otherwise raise questions regarding the acceptability of eliminating this Technical Specification Mode 2 scram requirement at relatively low power levels up to approximately 5% of rated reactor thermal power.

5.0 RESULTS

5.1 Steady-State Results

The initial steady-state run was made to establish the near full power conditions from which the model was "maneuvered" to reduce power in order to arrive at the desired conditions for the analysis. The second steady-state condition of interest was at the Mode 2 startup condition. This condition was achieved by performing a downpower maneuver using a combination of rod insertion and recirculation pump speed reduction, to achieve nearly shutdown conditions. Once the power was reduced in this manner, the feedwater enthalpy was controlled to minimize the core inlet subcooling and the turbine was tripped so that the steam flow could be controlled via the turbine bypass valve demand position table in the TRACG model. [[

]] The eigenvalue at the end of the downpower transient indicates the critical condition of the reactor.

Table 5.1

Sensed Power (% rated)	1.51
Vessel Dome Pressure (psia)	914
Recirc Pump Speed (% rated)	24.1
Recirc Pump Flow (% rated)	26.4
Jet Pump Flow (% rated)	16.3
Core Inlet Flow (% rated)	32.5
Core Inlet Subcooling (deg F)	1.88
Vessel Steam Flow (% rated)	2.77
Bypass Valve Demand (%)	8.5
Eigenvalue	0.9999944

5.2 Transient Results

5.2.1 Base Case – Comparison to Browns Ferry Test

The first transient simulates the Browns Ferry pressure perturbation test. There were actually two different pressure perturbation tests conducted at Browns Ferry as described in Reference 1. The first, as described on page 1 of Reference 1, opened one bypass valve rapidly (over about 0.1 sec), held it open for about 15 seconds, and then closed it rapidly (over about 0.1 sec). The second pressure perturbation test, mentioned very briefly on page 2 of Reference 1, slowly opened and closed two bypass valves. The purpose of performing the second test was to approximate the opening of one relief valve since one fully open bypass valve has a capacity of approximately 400,000 lbm/hr and the capacity of one relief valve is approximately 800,000 lbm/hr. The test report states that the rapid opening of the single bypass valve pressure perturbation test is the one of primary interest and, therefore, it is the one simulated in this analysis.

In the figures that follow, the data points for the Browns Ferry test were estimated from Figure 9 of the test report [Ref 1]. Table 5.2.1 provides the values used as "BF Test Data" points on the plots. Each x-interval on Figure 9 of Reference 1 corresponds to 1 second and, for purposes of this analysis, a test time of 60.0 seconds is defined as the point in time at which the bypass valve started opening in the test. The time scale is then adjusted in the following plots so that time 60.0 seconds matches the time from the TRACG runs at which the bypass was opened. This is called the "Problem Time" in Table 5.2.1. In addition, the statement in paragraph 1 on page 2 of Reference 1, which states that pressure dropped 16 psi in the first 3 seconds following the opening of the bypass valve, was used as the basis for assuming that each y-interval on Figure 9 corresponds to 1/10th of one inch. Furthermore, the level entries for the "test time" interval of 70 to 86 seconds in Table 5.2.1 are based on extrapolating the curve in an assumed reasonable manner because the curve in Figure 9 of Reference 1 goes off scale.

[[

]] The resulting turbine bypass valve steam flow is plotted in Figure 5.2.1-1.

Due to the sudden increase in steam flow, the vessel dome pressure drops and voids are formed in the core. This adds negative reactivity to the core causing power to drop as exhibited in Figure 5.2.1-2. The drop in vessel dome pressure also results in the core inlet fluid becoming saturated in less than 10 seconds. This is exhibited in Figure 5.2.1-3 and is consistent with the statements made in the test report (page 2 Reference 1) that boiling in the channels was postulated to begin 3 seconds after the vessel dome pressure began to fall off based on the observed drop in core flow. Although the model captures a similar rise and peak in core flow about 20 to 30 seconds after the initiation of the perturbation and then a subsequent fall in core flow, the model does not exhibit the same initial drop in core flow as was observed in the test. This is believed to be due to at least 2 factors. [[

]]

Regardless of the cause(s) of the short term core flow differences between the test and the model prediction, sensitivity calculations to assess the impact of the short term core flow difference showed this difference to not affect the well-behaved core responses.

Overall, the model does relatively well at capturing the initial response for the primary measurable parameter of interest that is available from the test data, i.e., the vessel dome pressure. Since the plant response is believed to be primarily driven by reactivity changes arising from void reactivity feedback, but void fraction is neither available from the test nor a measured plant process parameter, then the vessel dome pressure response provides the best indication of the model's capability to simulate the event. [[

]]

The level response in the three runs compares very favorably with the test data, especially with respect to timing of the peak level and the overall trend. [[

]]

The system response with the bypass demand decreased 20 seconds into the transient are plotted in Figures 5.2.1-7 through 5.2.1-12. The primary difference in this run is that with the reduced steam flow from the smaller bypass demand, the vessel dome pressure is allowed to increase resulting in an increase in power. The plots for this run are extended in time to show that the power eventually does peak and that the system response is well-behaved. The vessel dome pressure in this run eventually increases to the relief valve set point which, when opened, reduces the system pressure rapidly, increasing core voiding, and quickly reducing reactor power as a result.

Several conclusions are drawn from these comparisons. First, the TRACG model provides a reasonable, realistic response based on what are considered to be 'good' comparisons between the computer simulation and an actual 'similar' plant response. The comparisons are graded as 'good' considering the similar changes in the magnitude of the pressure and level response as well as the timing of peaks and valleys, i.e., the overall trend in the parameters. The comparisons are also graded as 'good' considering the differences in plant designs between the computer model and the test plant that might contribute to differences in the plant response. [[

]]

Table 5.2.1 Browns Ferry Test Data

Test Time (seconds)	Problem Time (seconds)	Dome Pressure (psia)	Level (inches)	Core Flow (Mlbm/hr)
[[				
				]]

Figure 5.2.1-1 Figure 5.2.1-2 Figure 5.2.1-3

[[ ]]

Figure 5.2.1-4 Figure 5.2.1-5 Figure 5.2.1-6

[[ ]]

Figure 5.2.1-7 Figure 5.2.1-8 Figure 5.2.1-9

[[ ]]

Figure 5.2.1-10 Figure 5.2.1-11 Figure 5.2.1-12

[[ ]]

5.2.2 Low Pressure Analysis

This analysis is used to evaluate the model's response to the pressure perturbation test at lower initial vessel dome pressures, as low as approximately 600 psia. Three different cases at different operating conditions are subjected to the pressure perturbation test. The first case is at an initial vessel dome pressure of approximately 750 psia with a core flow and recirculation pump flow of roughly 23% and 17% of rated flows, respectively. The second case is at an initial vessel dome pressure of approximately 600 psia with a core flow and recirculation pump flow of roughly 26% and 17% of rated flows, respectively. The third case is at an initial vessel dome pressure of approximately 650 psia with a core flow and recirculation pump flow of roughly 37% and 28% of rated flows, respectively. The first and second cases were developed from the same sequence of down pressure runs, but the third case (at 650 psia) specified a different recirculation pump speed in order to provide another core flow condition for the analysis.

A set of plots similar to those provided in Figures 5.2.1-1 through 5.2.1-6 are included as Figures 5.2.2-1 through 5.2.2-6 for the low pressure analysis for all three cases. The plots are arranged in order of decreasing initial vessel dome pressure from left to right. Thus, for each set of plots, the left-most plot labeled 'A' represents the 750 psia case, the middle plot labeled 'B' the 650 psia case, and the far-right plot labeled 'C' the 600 psia case. Table 5.2.2 summarizes several parameters for each case to facilitate comparison.

The system response is driven by a large number of interdependent factors. [[

]]

The overall response of the model continued to demonstrate that the system is stable and not susceptible to the pressure-power oscillations experienced by the old dual-cycle BWR plant.

NOTE: The Browns Ferry test level results were left superimposed on the plots of level simply because the model response was so similar to the test. Although it is understood that the initial vessel dome pressure conditions in these 3 computer runs are significantly different than the test conditions, the similarity of the level response does provide additional evidence about the physical reasonableness of the model.

Table 5.2.2

	Base Case	**Low Pressure Case 1**	**Low Pressure Case 2**	**Low Pressure Case 3**
Initial conditions				
Eigenvalue	[[			
Core Average Power (% rated)				
Vessel Dome Pressure (psia)				
Recirc Pump Speed (% rated)				
Recirc Pump Flow (% rated)				
Jet Pump Flow (% rated)				
Core Inlet Flow (% rated)				
Core Inlet Subcooling (deg F)				
Vessel Steam Flow (% rated)				
Bypass Valve Demand (%)				
Transient Parameters:				
Bypass Valve Flow Change (Mlb/hr)				
Core Avg Power Change (%)				
Core Void Fraction Change (%)				
Vessel Dome Pressure Change @ 15 sec (psi)				
Peak Level Swell/Shrink (in)				]]

Figure 5.2.2-1A

Figure 5.2.2-1B

Figure 5.2.2-1C

[[ ]]

Figure 5.2.2-2A

Figure 5.2.2-2B

Figure 5.2.2-2C

[[ ]]

Figure 5.2.2-3A

Figure 5.2.2-3B

Figure 5.2.2-3C

[[ ]]

Figure 5.2.2-4A

Figure 5.2.2-4B

Figure 5.2.2-4C

[[ ]]

Figure 5.2.2-5A

Figure 5.2.2-5B

Figure 5.2.2-5C

[[ ]]

Figure 5.2.2-6A

Figure 5.2.2-6B

Figure 5.2.2-6C

[[ ]]

5.2.3 Pressure-Induced Reactivity Analysis

This section of the analysis is intended to evaluate the system response to a rapid pressure increase caused by the fast closure of the bypass valves. It is referred to as the pressure-induced reactivity analysis because the increase in pressure will insert positive reactivity into the core due to the collapse of voids in the core region. Thus, instead of initially decreasing power as was done in the Browns Ferry test and in the computer simulations of Sections 5.2.1 and 5.2.2, the power will immediately increase and the system response to a positive reactivity addition will be addressed. Three cases are analyzed corresponding to the same initial conditions used for the low pressure analyses of Section 5.2.2. In all three cases, the bypass valves are fully closed in 0.1 seconds, held closed for 15 seconds, and then returned to their initial position over 0.1 seconds for the remainder of the transient. Table 5.2.3 summarizes key parameters for these runs and a similar set of plots to those provided in Figures 5.2.2-1 through 5.2.2-6 are included as Figures 5.2.3-1 through 5.2.3-6 for this analysis.

The closure of the bypass valves results in a rapid increase in pressure and a subsequent collapse of voids in the core. The positive reactivity added by the reduced voiding causes the power to increase rapidly and thereby mitigates the drop in core voids. [[

]] No power oscillations develop from the model's response to this significant pressure perturbation.

Table 5.2.3

	Low Pressure Reactivity Case 1	Low Pressure Reactivity Case 2	Low Pressure Reactivity Case 3
Initial conditions:			
Eigenvalue	[[		
Core Average Power (% rated)			
Vessel Dome Pressure (psia)			
Recirc Pump Speed (% rated)			
Recirc Pump Flow (% rated)			
Jet Pump Flow (% rated)			
Core Inlet Flow (% rated)			
Core Inlet Subcooling (deg F)			
Vessel Steam Flow (% rated)			
Bypass Valve Demand (%)			
Transient Parameters:			
Bypass Valve Flow Change (Mlb/hr)			
Core Avg Power Change (%)			
Core Void Fraction Change (%)			
Vessel Dome Pressure Change @ 15 sec (psi)			
Peak Level Swell/Shrink (in)			]]

Figure 5.2.3-1A

Figure 5.2.3-1B

Figure 5.2.3-1C

[[ ]]

Figure 5.2.3-2A

Figure 5.2.3-2B

Figure 5.2.3-2C

[[ ]]

Figure 5.2.3-3A Figure 5.2.3-3B Figure 5.2.3-3C

[[ ]]

Figure 5.2.3-4A Figure 5.2.3-4B Figure 5.2.3-4C

[[ ]]

Figure 5.2.3-5A

Figure 5.2.3-5B

Figure 5.2.3-5C

[[ ]]

Figure 5.2.3-6A

Figure 5.2.3-6B

Figure 5.2.3-6C

[[ ]]

5.2.4 Rod-Induced Reactivity Analysis

Up to this point, the perturbations imposed on the model have been initiated by pressure disturbances using the turbine bypass valves. This section of the analysis will examine the system response to perturbations initiated by reactivity changes due to rod movements. [[

]]

The first rod-induced reactivity cases analyzed are similar to the reactivity test performed at Browns Ferry. In these runs, a single rod near the center of the core was fully inserted and then fully withdrawn. Referring to Figure 4.1, the rod group 10 at location J-7, which corresponds to a single rod, was fully inserted over 1 second, held in that position for 60 seconds, and then fully withdrawn over 1 second. Table 5.2.4-1 summarizes key parameters for these 2 runs. Plots of interesting parameters are provided in Figures 5.2.4-1 through 5.2.4-8.

To determine the reactivity changes due to rod movement and void changes, the TRACG variables for average delayed neutron fraction (BETA), control rod reactivity (RCCL), and void reactivity (RCVD) of the TRACG model are used. The reactivity values are calculated by multiplying the applicable reactivity (rod or void) which is obtained from TRACG in units of '$' by the average delayed neutron fraction, then multiplying by 1e5 to convert to units of 'pcm', and finally subtracting the initial value of the applicable reactivity component at each time step.

[[

]] These reactivity changes are seen to have small effects on the core power, vessel dome pressure, level and so on which is consistent with the test data that showed this transient to be very mild.

The second rod-induced reactivity cases analyzed were intended to challenge the system more than the first cases. [[

]] Table 5.2.4-2 summarizes key parameters for these 2 runs. Plots of interesting parameters are provided in Figures 5.2.4-9 through 5.2.4-16.

[[

]] As expected, these reactivity changes are seen to have significant effects on the power, vessel dome pressure, level and so on; however, the system response was well-behaved in both cases and no signs of power oscillations in the system were apparent.

The third set of rod-induced reactivity cases analyzed was intended to provide a slow, continuous insertion of reactivity via a rod withdrawal. [[

]] In both cases, the system response to the slow withdrawal of the control rods is well-behaved with no signs of any pressure and power oscillations. Table 5.2.4-3 summarizes key parameters for these 2 runs and plots of interesting parameters are provided in Figures 5.2.4-17 through 5.2.4-23.

Table 5.2.4-1

	Rods Reactivity Case 1	Rods Reactivity Case 2
Initial conditions:		
Eigenvalue	[[	
Core Average Power (% rated)		
Vessel Dome Pressure (psia)		
Recirc Pump Speed (% rated)		
Recirc Pump Flow (% rated)		
Jet Pump Flow (% rated)		
Core Inlet Flow (% rated)		
Core Inlet Subcooling (deg F)		
Vessel Steam Flow (% rated)		
Bypass Valve Demand (%)		
Transient Parameters:		
Rod Reactivity Change (pcm)		
Void Reactivity Change (pcm)		
Core Avg Power Change (%)		
Avg Core Void Fraction Change (%)		
Vessel Dome Pressure Change (psi)		]]

Figure 5.2.4-1A

Figure 5.2.4-1B

[[ ]]

Figure 5.2.4-2A

Figure 5.2.4-2B

[[ ]]

Figure 5.2.4-3A

Figure 5.2.4-3B

[[ ]]

Figure 5.2.4-4A **Figure 5.2.4-4B**

[[ ]]

Figure 5.2.4-5A **Figure 5.2.4-5B**

[[ ]]

Figure 5.2.4-6A **Figure 5.2.4-6B**

[[ ]]

Figure 5.2.4-7A **Figure 5.2.4-7B**

[[ ]]

Figure 5.2.4-8A **Figure 5.2.4-8B**

[[ ]]

Table 5.2.4-2

	Rods Reactivity Case 3	**Rods Reactivity Case 4**
Initial conditions:		
Eigenvalue	[[	
Core Average Power (% rated)		
Vessel Dome Pressure (psia)		
Recirc Pump Speed (% rated)		
Recirc Pump Flow (% rated)		
Jet Pump Flow (% rated)		
Core Inlet Flow (% rated)		
Core Inlet Subcooling (deg F)		
Vessel Steam Flow (% rated)		
Bypass Valve Demand (%)		
Transient Parameters:		
Rod Reactivity Change (pcm)		
Void Reactivity Change (pcm)		
Core Avg Power Change (%)		
Avg Core Void Fraction Change (%)		
Vessel Dome Pressure Change (psi)		
Peak Level Swell/Shrink (in)		]]

Figure 5.2.4-9A **Figure 5.2.4-9B**

[[ ]]

Figure 5.2.4-10A **Figure 5.2.4-10B**

[[ ]]

Figure 5.2.4-11A **Figure 5.2.4-11B**

[[ ]]

Figure 5.2.4-12A **Figure 5.2.4-12B**

[[ ]]

Figure 5.2.4-13A **Figure 5.2.4-13B**

[[ ]]

Figure 5.2.4-14A **Figure 5.2.4-14B**

[[ ]]

Figure 5.2.4-15A **Figure 5.2.4-15B**

[[ ]]

Figure 5.2.4-16A **Figure 5.2.4-16B**

[[ ]]

Table 5.2.4-3

	Rods Reactivity Case 5	**Rods Reactivity Case 6**
Initial conditions:		
Eigenvalue	[[	
Core Average Power (% rated)		
Vessel Dome Pressure (psia)		
Recirc Pump Speed (% rated)		
Recirc Pump Flow (% rated)		
Jet Pump Flow (% rated)		
Core Inlet Flow (% rated)		
Core Inlet Subcooling (deg F)		
Vessel Steam Flow (% rated)		
Bypass Valve Demand (%)		]]

Figure 5.2.4-17A

Figure 5.2.4-17B

[[ ]]

Figure 5.2.4-18A

Figure 5.2.4-18B

[[ ]]

Figure 5.2.4-19A

Figure 5.2.4-19B

[[ ]]

Figure 5.2.4-20A Figure 5.2.4-20B

[[ ]]

Figure 5.2.4-21A Figure 5.2.4-21B

[[ ]]

Figure 5.2.4-22A Figure 5.2.4-22B

[[ ]]

Figure 5.2.4-23A **Figure 5.2.4-23B**

[[ ]]

5.2.5 Steam Volume Analysis

These cases are intended to evaluate the impact of operating in a bottled-up condition by closing the MSIVs so that the steam volume is reduced and the system response to pressure changes will be amplified. The sudden closure of the MSIVs results in a rapid decrease in void fraction and subsequent void reactivity addition as seen in Figures 5.2.5-3A & B and 5.2.5-7A & B. [[

]] The simulated system responds as expected. Overall, the simulated system response is very well-behaved and again exhibits no tendency to develop power oscillations as shown by the longer-term responses in Figures 5.2.5-1 through 5.2.5-6.

Table 5.2.5

	Steam Volume Case 1	**Steam Volume Case 2**
Initial conditions:		
Eigenvalue	[[	
Core Average Power (% rated)		
Vessel Dome Pressure (psia)		
Recirc Pump Speed (% rated)		
Recirc Pump Flow (% rated)		
Jet Pump Flow (% rated)		
Core Inlet Flow (% rated)		
Core Inlet Subcooling (deg F)		
Vessel Steam Flow (% rated)		
Bypass Valve Demand (%)		]]

Figure 5.2.5-1A **Figure 5.2.5-1B**

[[ ]]

Figure 5.2.5-2A **Figure 5.2.5-2B**

[[ ]]

Figure 5.2.5-3A **Figure 5.2.5-3B**

[[ ]]

Figure 5.2.5-4A **Figure 5.2.5-4B**

[[ ]]

Figure 5.2.5-5A **Figure 5.2.5-5B**

[[ ]]

Figure 5.2.5-6A **Figure 5.2.5-6B**

[[ ]]

Figure 5.2.5-7A **Figure 5.2.5-7B**

[[ ]]