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10 CFR 50.90

December 15, 2015

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

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

SUBJECT: License Amendment Request to Reduce Steam Dome Pressure Specified in the Reactor Core Safety Limits

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC), proposes a change to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively.

The proposed change revises PBAPS Technical Specifications (TS) Section 2.1.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. This change is consistent with the Nuclear Regulatory Commission (NRC) approved pressure range for the critical power correlations applied to the fuel type in use at PBAPS, Units 2 and 3.

The proposed change was identified as a result of GE Energy – Nuclear 10 CFR Part 21 Safety Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," issued on March 29, 2005, and is being submitted based on the results of subsequent GE analyses that were sponsored by the Boiling Water Reactor Owners Group.

EGC has concluded that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92.

The proposed change has been reviewed by the PBAPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

This amendment request contains no regulatory commitments.

Attachment 1 provides the evaluation of the proposed change. Attachment 2 provides a copy of the marked up TS pages that reflect the proposed change. Attachment 3 provides a copy of the marked up TS Bases pages that reflect the proposed change (information only).

EGC requests approval of the proposed amendment by December 15, 2016. Upon NRC approval, the amendment shall be implemented within 60 days of issuance.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the Commonwealth of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Stephanie J. Hanson at 610-765-5143.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of December 2015.

Respectfully,

James Barstow

James Barstow
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1. Evaluation of Proposed Change
2. Markup of Technical Specifications Pages
3. Markup of Technical Specifications Bases Pages (Information Only)

cc: USNRC Region I, Regional Administrator
USNRC Senior Resident Inspector, PBAPS
USNRC Project Manager, PBAPS
R. R. Janati, Bureau of Radiation Protection
S. T. Gray, State of Maryland

ATTACHMENT 1

EVALUATION OF PROPOSED CHANGES

Peach Bottom Atomic Power Station, Units 2 and 3

Docket Nos. 50-277 and 50-278

Subject: **License Amendment Request to Reduce Steam Dome Pressure Specified in the Reactor Core Safety Limits**

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC), proposes a change to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively.

The proposed change will revise PBAPS Technical Specifications (TS) Section 2.1.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. This change to TS Section 2.1.1 became necessary as a result of GE Energy - Nuclear 10 CFR Part 21 Reportable Condition Notification, Potential to Exceed Low Pressure Technical Specifications Safety Limit (Reference 1). This change is consistent with the Nuclear Regulatory Commission (NRC) approved pressure range for the critical power correlations applied to the fuel type in use at PBAPS, Units 2 and 3.

2.0 DETAILED DESCRIPTION

On March 29, 2005, GE Energy - Nuclear submitted a 10 CFR Part 21 notification (Reference 1) identifying that, as a result of applying improved methodologies for licensing basis transient analyses, the anticipated operational occurrence (AOO) Pressure Regulator Failure Maximum Demand (Open) (PRFO) had been identified as an event in which Reactor Core Safety Limit 2.1.1.1 could potentially be violated. GE has determined that this does not challenge the fuel cladding integrity. However, there is a potential vulnerability for the PRFO transient event to result in a condition in which TS SL 2.1.1.1 may be exceeded.

GE indicated that the approved model for licensing basis transient analysis had evolved from REDY, to ODYN, to TRACG. Reactor depressurization transients, such as PRFO, are non-limiting for fuel cladding integrity because critical power ratio (CPR) increases during the PRFO event, and are not typically included in the scope of cycle-specific reload evaluations. GE determined that REDY, ODYN, and TRACG all show the CPR increasing during the PRFO transient, and hence fuel cladding integrity not being challenged[1], and that the difference in reactor level swell predicted by REDY, versus ODYN and TRACG, can impact the predicted plant response to the PRFO.

GE indicated within the 10 CFR Part 21 notification letter that no clear compensatory action can be defined to appropriately mitigate this vulnerability, and since the condition does not challenge the physical barrier that the Safety Limit intends to protect (i.e., the fuel cladding integrity), there is no safety basis for a compensatory action. While this condition had been determined by GE to not involve an actual safety hazard, the potential for violation of a Reactor Core Safety Limit had been identified, and restoration to comply with the safety limit is required for the PRFO event. As a consequence, PBAPS is revising the reactor steam dome pressure TS Safety Limit to be consistent with the NRC approved pressure range of critical power correlations.

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[1] The Minimum Critical Power Ratio (MCPR) Safety Limit specified in Reactor Core Safety Limit 2.1.1.2 is established to protect fuel cladding integrity.

The proposed change to the PBAPS TS is summarized below:

1. The proposed change would revise the reactor steam dome pressure value in TS 2.1.1.1 and 2.1.1.2 from 785 psig to 685 psig for PBAPS, Units 2 and 3.

The marked up pages that reflect the proposed change are provided in Attachment 2 (TS pages) and Attachment 3 (TS Bases pages - information only).

3.0 TECHNICAL EVALUATION

Excessive thermal overheating of the fuel rod cladding can result in cladding damage and the release of fission products. In order to protect the cladding against thermal overheating due to boiling transition, the Safety Limits (SL) in PBAPS Technical Specifications (TS) 2.1.1 were established. Technical Specifications Safety Limits are specified to ensure that acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). Reactor Core SLs are set such that fuel cladding integrity is maintained and no fuel damage is calculated to occur if the SLs are not violated.

The Boiling Water Reactor (BWR) core is protected from the type of fuel failure that could occur during the Onset of Transition Boiling (OTB) by a combination of Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. Reactor Core Safety Limit 2.1.1.1 states that when the reactor steam dome pressure is less than 785 psig or when core flow is less than 10% of rated core flow, the reactor thermal power shall be less than or equal to 23% rated thermal power (RTP).

When reactor pressure and core flow are greater than or equal to the specified values in Reactor Core Safety Limits of TS 2.1.1.2, operation with a MCPR Safety Limit less than the values specified will be prohibited. The MCPR Safety Limit is established to ensure that at least 99.9 percent of the fuel rods in the core would not be expected to experience the onset of boiling transition. The SL of TS 2.1.1.1 was introduced to preclude the need for CPR calculations when reactor steam dome pressure is less than 785 psig or when core flow is less than 10% rated core flow by ensuring that reactor power would remain well below the fuel assembly critical power for the conditions in which CPR calculations are not performed (i.e., SL 2.1.1.1 limits thermal power to less than or equal to 23% RTP to ensure OTB conditions will not occur).

Reactor depressurization transients, such as Pressure Regulator Failure-Maximum Demand (Open) (PRFO), are non-limiting for fuel cladding integrity because CPR increases during the event, and they are not typically included in the scope of reload evaluations. Previous evaluations by GE using the REDY model predicted that reactor water level would swell during a PRFO transient; the depressurization would be terminated by a high level turbine trip. However, level swell is difficult to predict and the level swell portion of transient models have larger uncertainties than other portions of the transient models. Recent evaluations by GE with the improved transient models have determined that the reactor level swell may not be sufficient to reach the high level trip, in which case the depressurization could be terminated by Main Steam Isolation Valve (MSIV) closure at the low-pressure isolation setpoint (LPIS). Depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor steam dome pressure could decrease to below 785 psig for a few seconds while thermal power exceeds 23% RTP, which would exceed the conditions in TS SL 2.1.1.1. This issue was identified in the GE Part 21 Report (Reference 1).

In response to Reference 1, the BWR Owners' Group commissioned development of a methodology for plants to assess the adequacy of their current LPIS setting and to provide a set of recommendations for what actions should be taken based on the outcome of their assessment. The methodology is developed by analyzing a limiting plant, assessing uncertainties, and determining a method to conservatively scale the limiting plant's results to other plant configurations and operating flexibility options through sensitivity studies. The scaling methodology is applied to an example plant to demonstrate its adequacy. Additionally, a parametric study using a 720 psig LPIS setting with various plant configurations is provided.

The methodology identified in NEDC-33743P "BWR Owners' Group Reload Analysis and Core Management Committee SC05-03 Analysis Report," Revision 0 (Reference 2) was used to assess adequacy of PBAPS current LPIS for the issue identified in SC05-03 (Reference 1). The assessment determined that the LPIS analytical limit at PBAPS is sufficient to preclude steam dome pressure from falling below 685 psig (700 psia) while above 23% RTP during a PRFO event.

PBAPS current fuel design consists of GNF2 fuel. GE utilizes the GEXL correlation to perform CPR calculations for the fuel type in use at Peach Bottom. The lower bound limit of 685 psig (700 psia) for the GEXL17 correlation is documented and justified in GE NEDC-33292P "GEXL17 correlation for GNF2 Fuel" (Reference 3). This lower bound limit is discussed in NEDC-33292P and is referenced in NEDC-33270P (Reference 4). NEDC-33270P was submitted to the NRC as part of Amendment No. 33 to NEDE-24011-P. NEDE-24011-P Amendment No. 33 was approved by the NRC and incorporated into Revision 17 of NEDE-24011-P-A (Reference 5). Therefore, the use of 685 psig (700 psia) as lower bound limit for GNF2 fuel has been approved by the NRC for use per NEDE-24011-P-A by reference.

Reference 6 (MELLLA+) discusses the limitations and conditions associated with applications of GE methods to expanded operating domains. Appendix B of Reference 6 (MELLLA+) discusses the limitations and conditions associated with applications of MELLLA+ operating domain. None of the limitations and conditions are associated with lower bound limit for CPR correlations.

Use of 685 psig (700 psia) as steam dome pressure limit for TS 2.1.1.1 and TS 2.1.1.2 is supported by the CPR correlations in use for PBAPS. The minimum steam dome pressure resulting from a PRFO event is demonstrated to be above 685 psig (700 psia) using Reference 2 methods. Revising the Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 reactor steam dome pressure from 785 to 685 psig resolves the 10 CFR Part 21 condition concerning the potential to violate Reactor Core Safety Limit 2.1.1.1 during a PRFO transient reported in Reference 1. If Exelon decides to switch to a different fuel design from what is currently in use in the PBAPS reactor core, the CPR correlation will be reviewed as part of the normal fuel design change and reload licensing processes. If the CPR correlation for the new fuel design has a lower bound pressure which is higher than the limit specified in the Technical Specifications, then a LAR will be submitted for staff review and approval. If the CPR correlation has a lower bound pressure which is lower than the Technical Specifications limit, then no LAR will be required since the Technical Specifications would set a conservative lower bound.

Results of the above Exelon evaluations show that the Extended Power Uprate LPIS setting at PBAPS is adequate to prevent reactor pressure from falling below 685 psig (700 psia) while power is above 23% RTP during a PRFO event. The CPR correlation currently in use at PBAPS supports lower bound pressure of 685 psig (700 psia).

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The following regulatory requirements have been considered:

- Title 10 of the Code of Federal Regulations (10 CFR), Section 50.36, "Technical specifications," in which the Commission established its regulatory requirements related to the contents of the TS. 10 CFR 50.36(c) requires that the TS include, among other things, items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. In addition, 10 CFR 50.36 states that the TS will include Safety Limits for nuclear reactors which are stated to be "limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

The applicable 10 CFR Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, was considered as follows:

- Criterion 10 - Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

  The purpose of the safety limit is to ensure that specified acceptable fuel design limits are not exceeded during steady state operation and analyzed transients. The fuel cladding is one of the physical barriers that separate the radioactive materials from the environment. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Fuel cladding perforations can result from thermal stresses, which can occur from reactor operation significantly above design conditions. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel cladding damage could occur. The reactor core safety limits are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur due to onset of transition boiling if the safety limits are not exceeded.

  In addition, the reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The reactor core components consist of fuel assemblies, control rods, in-core ion chambers and related items. The fuel is designed to provide high integrity over a complete range of power levels including transient conditions. As described above, the PBAPS TS SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and AOOs. Reactor core SLs are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur if the SLs are not exceeded.

The proposed TS change revises the reactor steam dome pressure stated in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 to remove the potential to violate Reactor Core Safety Limit 2.1.1.1 during a Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient.

As long as the core pressure and flow are within the range of validity of the specified critical power correlation the proposed reactor steam dome pressure change to Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 will continue to ensure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition. This satisfies the requirements of GDC 10 regarding acceptable fuel design limits and continues to assure that the underlying criteria of the safety limit is met. Based on this, there is reasonable assurance that the health and safety of the public is unaffected.

4.2 Precedence

The NRC has previously reviewed requests for TS changes in support of resolving the GE Part 21 concern similar to this proposed amendment request for PBAPS as documented in the following approved amendments:

1. On March 11, 2013, Northern States Power Company – Minnesota, submitted a License Amendment Request proposing to reduce the reactor steam dome pressure specified in Reactor Core Safety Limit Specification 2.1.1 (Reference 7). The NRC approved amendment 185 for the Monticello Nuclear Generating Plant on November 25, 2014 (Reference 8).

2. On March 24, 2014, Southern Nuclear Operating Company, submitted an amendment request to revise the Edwin I. Hatch Plant Units 1 and 2 TS Section 2.1.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 (Reference 9). The NRC completed their review and issued amendments 269 and 213 on October 20, 2014 (Reference 10).

3. On May 28, 2013, Entergy Operations, Inc., submitted an amendment request to revise the River Bend Station TS Section 2.1.1 to reflect a lower reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 (Reference 11). The NRC completed their review and issued amendment 182 on December 11, 2014 (Reference 12).

4. On October 8, 2013, Entergy Nuclear Operations, Inc., proposed an amendment to modify the James A FitzPatrick Nuclear Power Plant TS to reduce the reactor pressure associated with the Reactor Core Safety Limit in TS 2.1.1.1 and TS 2.1.1.2 (Reference 13). The NRC completed their review and issued amendment 309 on February 9, 2015 (Reference 14).

4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (EGC), proposes a change to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively.

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 change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No. The proposed change to the reactor steam dome pressure in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 does not alter the use of the analytical methods used to determine the safety limits that have been previously reviewed and approved by the NRC. The proposed change is in accordance with an NRC approved critical power correlation methodology, and as such, maintains required safety margins. The proposed change does not adversely affect accident initiators or precursors, nor does it alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained.

The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not require any physical change to any plant SSCs nor does it require any change in systems or plant operations. The proposed change is consistent with the safety analysis assumptions and resultant consequences.

Lowering the value of reactor steam dome pressure in the TS has no physical effect on plant equipment and therefore, no impact on the course of plant transients. The change is an analytical exercise to demonstrate the applicability of correlations and methodologies. There are no known operational or safety benefits.

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

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

Response: No. The proposed reduction in the reactor dome pressure safety limit from 785 psig to 685 psig is a change based upon previously approved documents and does not involve changes to the plant hardware or its operating characteristics. As a result, no new failure modes are being introduced. There are no hardware changes nor are there any changes in the method by which any plant systems perform a safety function. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change.

The proposed change does not introduce any new accident precursors, nor does it involve any physical plant alterations or changes in the methods governing normal plant operation. Also, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis.

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

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

Response: No. The margin of safety is established through the design of the plant structures, systems, and components, and through the parameters for safe operation and setpoints for the actuation of equipment relied upon to respond to transients and design basis accidents. Evaluation of the 10 CFR Part 21 condition by General Electric determined that since the Minimum Critical Power Ratio improves during the PRFO transient, there is no decrease in the safety margin and therefore there is no threat to fuel cladding integrity. The proposed change in reactor steam dome pressure supports the current safety margin, which protects the fuel cladding integrity during a depressurization transient, but does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. The change does not alter the behavior of plant equipment, which remains unchanged.

The proposed change to Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 is consistent with and within the capabilities of the applicable NRC approved critical power correlation for the fuel design in use at PBAPS Units 2 and 3. No setpoints at which protective actions are initiated are altered by the proposed change. The proposed change does not alter the manner in which the safety limits are determined. This change is consistent with plant design and does not change the TS operability requirements; thus, previously evaluated accidents are not affected by this proposed change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, EGC concludes that the proposed amendment presents no 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.

4.4 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

A review 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 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. Letter from Jason Post (GE Energy – Nuclear) to U.S. NRC, "10CFR21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," dated March 29, 2005 (ADAMS Accession No. ML050950428).

2. NEDC-33743P, Revision 0, "BWR Owners' Group Reload Analysis and Core Management Committee SC05-03 Analysis Report," GE Hitachi Nuclear Energy, April 2012.

3. NEDC-33292P, Revision 3, "GEXL17 correlation for GNF2 Fuel," Global Nuclear Fuel, June 2009.

4. NEDC-33270P, Revision 3, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," Global Nuclear Fuel, March 2010.

5. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Global Nuclear Fuel.

6. NEDC-33720P, Revision 0, "Safety Analysis Report for Peach Bottom Atomic Power Station Units 2&3, Maximum Extended Load Line Limit Analysis Plus", GE Hitachi Nuclear Energy, September 2014.

7. Letter from John C. Grubb (Northern States Power Company – Minnesota) to U.S. NRC, "License Amendment Request: Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety limits," dated March 11, 2013 (ADAMS Accession No. ML13074A811).

8. Letter from Terry A. Beltz (U.S. NRC) to Karen D. Fili (Northern States Power Company – Minnesota), "Monticello Nuclear Generating Plant – Issuance of Amendment to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits (TAC No. MF1054)," dated November 25, 2014 (ADAMS Accession No. ML14281A318).

9. Letter from C. R. Pierce (Southern Nuclear Operating Company) to U.S. NRC, "License Amendment Request to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits," dated March 24, 2014 (ADAMS Accession No. ML14084A201).

10. Letter from Robert Martin (U.S. NRC) to C. R. Pierce (Southern Nuclear Operating Company), "Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Issuance of Amendments Reducing the Reactor Steam Dome Pressure in the Reactor Core Safety Limits (TAC Nos. MF3722 and MF3723)," dated October 20, 2014 (ADAMS Accession No. ML14276A634).

11. Letter from Eric W. Olson (Entergy Operations, Inc.) to U.S. NRC, "License Amendment Request Changes to Technical Specification 2.1.1, 'Reactor Core SLs'," dated May 28, 2013.

12. Letter from Alan Wang (U.S. NRC) to Vice President, Operations (Entergy Operations, Inc.), "River Bend Station, Unit 1 – Issuance of Amendment Re: Technical Specification 2.1.1, 'Reactor Core SLs' (TAC No. MF1948)," dated December 11, 2014.

13. Letter from Lawrence M. Coyle (Entergy Nuclear Operations, Inc.) to U.S. NRC, "Application to Revise Technical Specifications for Technical Specification Low Pressure Safety Limit," dated October 8, 2013.

14. Letter from Douglas V. Pickett (U.S. NRC) to Vice-President, Operations (Entergy Nuclear Operations, Inc.), "James A FitzPatrick Nuclear Power Plant – Issuance of Amendment Re: Application to Revise Technical Specifications for Technical Specification Low Pressure Safety Limit (TAC No. MF2897)," dated February 9, 2015.

ATTACHMENT 2

Markup of Technical Specifications Pages

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
Docket Nos. 50-277 and 50-278

License Amendment Request to
Reduce Steam Dome Pressure Specified in the Reactor Core Safety Limits

Unit 2 TS Pages

2.0-1

Unit 3 TS Pages

2.0-1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785~~ psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

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2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.14 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785~~ psig or core flow < 10% rated core flow:

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THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

ATTACHMENT 3

Markup of Technical Specifications Bases Pages
(Information Only)

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
Docket Nos. 50-277 and 50-278

License Amendment Request to
Reduce Steam Dome Pressure Specified in the Reactor Core Safety Limits

Unit 2 TS Bases Pages

B 2.0-2
B 2.0-3
B 2.0-6
B 3.3-147

Unit 3 TS Bases Pages

B 2.0-2
B 2.0-3
B 2.0-6
B 3.3-148

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormal operational transients. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ ~~785~~ 685 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

The pressure drop in the bypass region is essentially all elevation head with a value > 4.5 psi; therefore, the core pressure drop at low power and flows will always be > 4.5 psi. At power, the static head inside

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

the bundle is less than the static head in the bypass region because the addition of heat reduces the density of the water. At the same time, dynamic head loss in the bundle will be greater than in the bypass region because of two phase flow effects. Analyses show that this combination of effects causes bundle pressure drop to be nearly independent of bundle power when bundle flow is 28 X 10^3 lb/hr and bundle pressure drop is 3.5 psi. Because core pressure drop at low power and flows will always be > 4.5 psi, the bundle flow will be > 28 X 10^3 lb/hr.

Full scale ATLAS test data taken at pressures from 14.7 psia (0 psig) to 800 psia (785 psig) indicate that the fuel assembly critical power with bundle flow at 28 X 10^3 lb/hr is approximately 3.35 MWt. This is equivalent to a THERMAL POWER > 50% RTP even when design peaking factors are considered. Therefore, a THERMAL POWER limit of 23% RTP ~~prevents any bundle from exceeding critical power and is a conservative limit when reactor pressure < 785 psig.~~

for reactor pressure < 685 psig is conservative. Additional information on low flow conditions is available in Reference 4.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore,

(continued)

BASES

REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

2. 10 CFR 100.

3. 10 CFR 50.67.

4. SIL No. 516 Supplement 2, January 19, 1996.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level—Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure—Low

during the depressurization transient in order to maintain reactor steam dome pressure > 685 psig. The MSIV closure

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs ~~prior to pressure decreasing below 785 psig, which~~ results in a scram ~~due to MSIV closure~~, thus reducing reactor power to < 23% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

BACKGROUND (continued)	Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormal operational transients. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

<u>2.1.1.1 Fuel Cladding Integrity</u>

GE critical power correlations are applicable for all critical power calculations at pressures ≥ ~~785~~ 685 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> The pressure drop in the bypass region is essentially all elevation head with a value > 4.5 psi; therefore, the core pressure drop at low power and flows will always be > 4.5 psi. At power, the static head inside

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

the bundle is less than the static head in the bypass region because the addition of heat reduces the density of the water. At the same time, dynamic head loss in the bundle will be greater than in the bypass region because of two phase flow effects. Analyses show that this combination of effects causes bundle pressure drop to be nearly independent of bundle power when bundle flow is 28 X 10^3 lb/hr and bundle pressure drop is 3.5 psi. Because core pressure drop at low power and flows will always be > 4.5 psi, the bundle flow will be > 28 X 10^3 lb/hr.

Full scale ATLAS test data taken at pressures from 14.7 psia (0 psig) to 800 psia (785 psig) indicate that the fuel assembly critical power with bundle flow at 28 X 10^3 lb/hr is approximately 3.35 MWt. This is equivalent to a THERMAL POWER > 50% RTP even when design peaking factors are considered. Therefore, a THERMAL POWER limit of 23% RTP ~~prevents any bundle from exceeding critical power and is a conservative limit when reactor pressure < 785 psig.~~ for reactor pressure < 685 psig is conservative. Additional information on low flow conditions is available in Reference 5.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore,

(continued)

BASES

REFERENCES

1. DELETED

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

3. 10 CFR 100.

4. 10 CFR 50.67.

5. SIL No. 516 Supplement 2, January 19, 1996.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low (Level 1) (continued)

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1.b. Main Steam Line Pressure—Low

during the depressurization transient in order to maintain reactor steam dome pressure > 685 psig. The MSIV closure

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs ~~prior to pressure decreasing below 785 psig, which~~ results in a scram ~~due to MSIV closure~~, thus reducing reactor power to < 23% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

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