NEXTera ENERGY DUANE ARNOLD

August 6, 2015

NG-15-0048
10 CFR 50.90

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

Duane Arnold Energy Center
Docket No. 50-331
Renewed Facility Operating License No. DPR-49

License Amendment Request (TSCR-153) to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits

References:

1. GE Energy-Nuclear, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," MFN 05-021, March 29, 2005 (ML050950428)

In accordance with the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), NextEra Energy Duane Arnold, LLC (hereafter, NextEra Energy Duane Arnold) is submitting a request for an amendment to the Technical Specifications (TS) for Duane Arnold Energy Center (DAEC).

The proposed amendment resolves a 10 CFR Part 21 condition concerning a potential to momentarily violate Reactor Core Safety Limit 2.1.1.1 during Pressure Regulator Failure Maximum Demand (Open) transient reported in Reference 1.

Attachment 1 provides an evaluation of the proposed changes. Attachment 2 provides marked-up pages of existing TS to show the proposed changes. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides the marked-up TS Bases pages for information only. There are no revisions to existing Regulatory Commitments.

NextEra Energy Duane Arnold requests approval of the proposed license amendment within one year, with the amendment being implemented within 60 days of its receipt.

In accordance with 10 CFR 50.91(a)(1), "Notice for Public Comment," the analysis about the issue of no significant hazards consideration using the standards in 10 CFR 50.92 is being provided to the Commission.

A001
NRR

Rec'd @ OPC
on 9/2/15

In accordance with 10 CFR 50.91(b)(1), "Notice for Public Comment; State Consultation," a copy of this application and its reasoned analysis about no significant hazards considerations is being provided to the designated State of Iowa official.

The DAEC Onsite Review Group has reviewed the proposed license amendment request.

If you have any questions or require additional information, please contact J. Michael Davis at 319-851-7032.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 6, 2015.

T. A. Vehec
Vice President, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

Attachments: As stated

cc: Regional Administrator, USNRC, Region III,
Project Manager, USNRC, Duane Arnold Energy Center
Resident Inspector, USNRC, Duane Arnold Energy Center
A. Leek (State of Iowa)

ATTACHMENT 1 to NG-15-0048

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-153) TO REDUCE THE REACTOR STEAM DOME PRESSURE SPECIFIED IN THE REACTOR CORE SAFETY LIMITS

EVALUATION OF PROPOSED CHANGES

7 pages follow

1.0 DESCRIPTION

On March 29, 2005, General Electric (GE) submitted a Safety Communication (SC 05-03) in accordance with 10 CFR 21.21(d) (Reference 9.1). GE identified an anticipated operational occurrence (AOO), the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient, that could result in a condition in which Safety Limit (SL) 2.1.1.1 may be exceeded.

Accordingly, pursuant to 10 CFR 50.90, NextEra Energy Duane Arnold, LLC (NextEra Energy Duane Arnold) hereby requests an amendment to Duane Arnold Energy Center (DAEC) Technical Specifications (TS). The requested amendment would revise the reactor dome pressure from 785 psig to 685 psig in TS SLs 2.1.1.1 and 2.1.1.2 to resolve the potential to violate these limits during a PRFO transient.

2.0 PROPOSED CHANGES

The proposed changes reduce the reactor steam dome pressure specified in TS SLs 2.1.1.1 and 2.1.1.2 from 785 psig to 685 psig. The TS SLs 2.1.1.1 and 2.1.1.2 would then read:

- 2.1.1.1 Fuel Cladding Integrity - With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

  THERMAL POWER shall be ≤ 21.7% RTP.

- 2.1.1.2 MCPR - With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

  MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

A marked-up copy of the proposed changes to the TS SLs 2.1.1.1 and 2.1.1.2 is provided in Attachment 2. Proposed revisions to the TS Bases are also included for information only in Attachment 4. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program upon receipt of the NRC approved License Amendment.

3.0 HISTORY

On March 29, 2005, GE submitted Reference 9.1, which informed affected licensees that recent evaluations with improved transient models have determined that the reactor level during a PRFO transient may not be sufficient to reach the high level trip, in which case the depressurization could be terminated by Main Steam Isolation Valve 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 21.7% of rated, which would violate the conditions in DAEC TS SL 2.1.1.1.

GE indicated within Reference 9.1 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 limits is required for the PRFO transient. GE advanced fuel designs have an NRC approved critical power correlation with a lower-bound pressure significantly below the 785 psig reactor steam dome pressure specified in DAEC TS SLs 2.1.1.1 and 2.1.1.2. NextEra Energy Duane Arnold proposes to use this fact and reduce the reactor steam dome pressure specified in TS SLs 2.1.1.1 and 2.1.1.2 from 785 psig to 685 psig consistent with NRC approved pressure range of critical power correlations for DAEC fuel designs.

4.0 BACKGROUND

A discussion providing background on TS 2.1.1, "Reactor Core SLs," and a summary of the PRFO transient scenario considering the change in computer analysis codes is provided below.

4.1 Background on the Reactor Core Safety Limits

TS Safety Limits are specified to ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences. The Reactor Core Safety Limits are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur if the Safety Limits are not exceeded.

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 when the reactor steam dome pressure is less than 785 psig or when core flow is less than 10% of rated core flow, reactor thermal power shall be less than or equal to 21.7% rated thermal power (RTP). When reactor pressure and core flow are greater than these specified values, Reactor Core Safety Limit 2.1.1.2 prohibits operation with a minimum critical power ratio (MCPR) Safety Limit less than the values specified to prevent fuel cladding damage that could occur when a fuel assembly experiences the OTB.

As discussed in Section B 2.1.1 of the TS Bases, for operation at low pressures or low flows, such as during startup, an alternate basis is used to provide fuel cladding integrity protection. Reactor Core Safety Limits 2.1.1.1 precludes the need for critical power ratio (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 at which CPR calculations are not performed (i.e., Safety Limit 2.1.1.1 limits thermal power to less than or equal to 21.7% RTP to ensure OTB conditions will not occur).

4.2 Background for Pressure Regulator Failure Maximum Demand (Open) Transient Analysis

The GE Part 21 report describes a revised transient analysis scenario for the PRFO event. A change in the predicted series of events for this transient was identified based upon a change in computer codes and the predicted results of this event.

Previous evaluations using the REDY methodology indicated the transient would be terminated by direct turbine trip and subsequent reactor scram resulting from the reactor water level swell following the event. Specifically, for the postulated event, the pressure regulator system fails in such a manner that a demand occurs to open the turbine steam admission valves, i.e., turbine stop valves (TSVs), turbine control valves, and turbine bypass valves. As a result, the reactor depressurization causes the formation of voids within the reactor core. The core voiding increases the reactor water level until the level reaches the main turbine trip (level) setpoint. The turbine trips, in turn sending a direct signal (via the TSV position switches) to the reactor protection system (RPS) resulting in the reactor automatically shutting down, terminating the transient.

A somewhat different series of events is predicted when the event was analyzed with improved transient methods. The transient occurs as before and the reactor depressurizes; however, the reactor level does not swell to the level setpoint to cause a main turbine trip. Level swell is difficult to predict and the level swell portion of transient models have larger uncertainties than other portions of the transient models. In this case the depressurization could be terminated by Main Steam Isolation Valve (MSIV) closure at the low-pressure isolation setpoint (LPIS). (The Main Steam Line Pressure – Low Function (Function 1.b in TS Table 3.3.6.1-1) for DAEC corresponds to the LPIS in the 10 CFR Part 21 notification.) This results in the transient not being terminated as quickly as the earlier methods predicted. Reactor depressurization continues to occur until the pressure decreases to the MSIV closure (in MODE 1) containment isolation signal setpoint. The MSIV closure is a direct input, via position switches, to the RPS. The reactor scrams and the transient is terminated.

However, under this series of events, the delay in termination of the transient introduces the possibility for reactor pressure to decrease below the 785 psig TS limit while reactor power is still greater than 21.7% RTP. Depending upon the plant-specific response to a PRFO event, including the value of the LPIS and the closure rate for the MSIV, reactor steam dome pressure could decrease to below 785 psig for a few seconds while thermal power exceeds 21.7% RTP, which would violate the conditions in Reactor Core Safety Limit 2.1.1.1. This indicates that Reactor Core Safety Limit 2.1.1.1 is overly conservative with respect to this event because during this event CPR continues to increase and therefore does not threaten fuel cladding integrity. The pressure decrease, though, could result in violating the value specified in the safety limit specification, while having no actual safety significance.

5.0 TECHNICAL ANALYSIS

Technical Specification 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). The purpose of Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 is to protect fuel cladding integrity. The fuel cladding integrity safety limit (MCPR Safety Limit) is defined as the CPR in the limiting fuel assembly for more than 99.9% of the fuel rods in the core are expected to avoid transition boiling, considering the power distribution within the core and all uncertainties. The safety limit is set such that no significant fuel damage is calculated to occur if the limit is not violated. It is determined using a statistical model that combines the uncertainties in operating parameters and procedures used to calculate critical power.

The probability of the occurrence of OTB is determined using approved critical power correlations. Each fuel vendor has developed correlations valid over specified pressure and flow ranges (mass flow rates) that are approved by the NRC. The critical power correlations for some advanced fuel designs have received NRC approval down to a lower pressure than those approved previously. The lower-bound of the extended pressure ranges for these advanced fuel designs can be used to establish a lower reactor steam dome pressure than the 785 psig value currently specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. NextEra Energy

Duane Arnold proposes to utilize the fact that the GE14 and GNF2 fuel that comprise the DAEC core, utilize critical power correlations that have an approved pressure range from 700 to 1400 psia. The GE 14 fuel design was introduced during DAEC Operating Cycle 18. The GNF2 fuel design was introduced during DAEC Operating Cycle 24. DAEC is currently in operating Cycle 25. In accordance with 10 CFR 50.59, only fuel which has an NRC approved CPR correlation with a lower-bound pressure less than or equal to the reactor steam dome pressure specified in the safety limit may be loaded into the core. 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 reported 10 CFR Part 21 condition concerning the potential to violate Reactor Core Safety Limit 2.1.1.1 during a PRFO transient by offering a greater pressure margin for a PRFO transient than what is currently available. DAEC UFSAR Section 15.1.7.1 provides the plant response to a PRFO transient. 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.

6.0 <u>REGULATORY SAFETY ANALYSIS</u>

6.1 <u>NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION</u>

NextEra Energy Duane Arnold has evaluated the proposed changes to the Technical Specifications (TS) using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

Description of Amendment Request: The proposed changes reduce the reactor steam dome pressure from 785 psig to 685 psig in TS Safety Limits (SLs) 2.1.1.1 and 2.1.1.2.

Basis for proposed no significant hazards determination: As required by 10 CFR 50.91(a), the NextEra Energy Duane Arnold analysis of the issue of no significant hazards consideration is presented 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 from 785 psig to 685 psig in TS SLs 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.

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 change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. 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 impose any new or different requirements or eliminate any existing requirements. The proposed 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

   Margin of safety is related to confidence in the ability of the fission product barriers (fuel cladding, reactor coolant system, and primary containment) to perform their design functions during and following postulated accidents. Evaluation of the 10 CFR Part 21 condition by General Electric determined that there was no decrease in the safety margin, the Minimum Critical Power Ratio improves during the transient, and therefore is not a threat to fuel cladding integrity.

   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, and thus continues to ensure that valid critical power calculations are performed. 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.

6.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

10 CFR 50.36, "Technical Specifications," provides the regulatory requirements for the content required in the TS. As stated in 10 CFR 50.36(c)(1)(i)(A),

> Safety limits for nuclear reactors are 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. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission.

DAEC UFSAR Section 3.1, "Conformance to AEC General Design Criteria for Nuclear Power Plants," provides an evaluation of the design basis of DAEC against Appendix A of 10 CFR 50 effective May 21, 1971 and subsequently amended on July 7, 1971. The applicable AEC General Design Criteria (GDC) is Criterion 10, "Reactor Design," which states, "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." NextEra Energy Duane Arnold has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. 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 99.9% of the fuel rods in the core are expected to avoid the onset of boiling transition. This satisfies the requirements of Criterion 10 regarding specified 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, following approval of this TS change is unaffected.

7.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment of an operating license for a facility requires no environmental assessment, if the operation of the facility in accordance with the proposed amendment does not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) result in a significant increase in individual or cumulative occupational radiation exposure. NextEra Energy Duane Arnold has reviewed this license amendment request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination is as follows.

Basis

This change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in the 10 CFR 50.92 evaluation, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed amendment does not change or modify the design or operation of any plant systems, structures, or components. The proposed amendment does not affect the

amount or types of gaseous, liquid, or solid waste generated onsite. The proposed amendment does not directly or indirectly affect effluent discharges.

3. The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed amendment does not change or modify the design or operation of any plant systems, structures, or components. The proposed amendment does not directly or indirectly affect the radiological source terms.

8.0 PRECEDENT

This License Amendment Request is similar to a License Amendment Request approved by letter dated October 20, 2014 (ML14276A634), "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)," and another License Amendment Request approved by letter dated November 25, 2014 (ML14281A318), "Monticello Nuclear Generating Plant – Issuance of Amendment to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits (TAC NO. MF1054)."

9.0 REFERENCES

9.1 GE Energy-Nuclear, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," MFN 05-021, March 29, 2005 (ML050950428)

ATTACHMENT 2 to NG-15-0048

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-153)
TO REDUCE THE REACTOR STEAM DOME PRESSURE SPECIFIED IN THE REACTOR CORE SAFETY LIMITS

PROPOSED TECHNICAL SPECIFICATION CHANGES
(MARKUP COPY)

1 page follows

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

THERMAL POWER shall be ≤ 21.7% RTP. 685

2.1.1.2 MCPR – 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 ≥ 1.12 for single recirculation loop operation.

2.1.1.3 Reactor Vessel Water Level – Reactor vessel water level shall be greater than 15 inches above the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1335 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 Fully insert all insertable rods.

ATTACHMENT 3 to NG-15-0048

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-153)
TO REDUCE THE REACTOR STEAM DOME PRESSURE SPECIFIED IN THE REACTOR CORE SAFETY LIMITS

REVISED TECHNICAL SPECIFICATION PAGE

1 page follows

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 Fuel Cladding Integrity – With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 21.7% RTP.

2.1.1.2 MCPR – With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.1.3 Reactor Vessel Water Level – Reactor vessel water level shall be greater than 15 inches above the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1335 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 Fully insert all insertable rods.

ATTACHMENT 4 to NG-15-0048

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-153)
TO REDUCE THE REACTOR STEAM DOME PRESSURE SPECIFIED IN THE REACTOR CORE SAFETY LIMITS

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES
(FOR INFORMATION ONLY)

3 pages follow

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 ECCS initiation setpoints higher than the SL provides margin to the SL but is independent of the SL.

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 the 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:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 43% RTP. Thus, a THERMAL POWER limit of 21.7% RTP for reactor pressure < ~~785~~ 685 psig is conservative.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant 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 transition boiling 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, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid transition boiling, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

For SLO, the SLMCPR is greater to account for the increased uncertainties.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Main Steam Line Isolation

1.a. Reactor Vessel Water Level — Low Low Low

Low Reactor Pressure Vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level — Low Low Low Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level — Low Low Low Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Reactor Vessel Water Level - Low Low Low supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level — Low Low Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level — Low Low Low Allowable Value is chosen to be the same as the ECCS Reactor Vessel Water Level - Low Low Low 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 the Group 1 valves.

1.b. Main Steam Line Pressure — Low

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. 2). 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
685

(continued)