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10 CFR 50.90
10 CFR 50.91

102-04425-CDM/SAB/RKR
March 31, 2000

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Reference: Letter 102-04293-JML/SAB/RKR, dated May 26, 1999, from J. M. Levine, APS, to NRC, "Request for Amendment to Technical Specification 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating."

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN 50-528/529/530
Response to NRC Request for Additional Information and Revised Request for Amendment to Technical Specification 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating

In the referenced letter, Arizona Public Service Company (APS) requested an amendment to Technical Specification 3.3.1, Reactor Protective System (RPS) Instrumentation - Operating, for each Palo Verde Nuclear Generating Station (PVNGS) Unit. In phone calls on August 17, 1999 and October 5, 1999 the NRC and APS staffs discussed the specific methodology used in the reactor coolant pump (RCP) sheared shaft event analysis affected by the amendment request. At the end of the October 5 phone call, the NRC staff requested that APS formally submit the information discussed in the phone calls. Enclosure 2 provides the requested information.

RCP Sheared Shaft Analysis

In the preparation of Enclosure 2, a review of the guidance provided by NRC Office Letter Number 803, "Technical Specification Review Procedures," was performed. APS engineering reviewed the methodology used for the sheared shaft analysis discussed in the amendment request to determine if the methodology was different than the methodology described in Updated Final Safety Analysis (UFSAR) section 15.3.4, "Reactor Coolant Pump Shaft Break with Loss of Offsite Power." This review identified that an assumption used in the dose calculation portion of the analysis had been changed. The UFSAR analysis methodology assumes that the atmospheric dump valves (ADVs) are opened 30 minutes after the reactor coolant pump shaft breaks and that one ADV is stuck open for the duration (90 minutes) of the analyzed event.

A001

The review found that the supporting dose consequence analysis for the amendment request assumed that the stuck open ADV was manually closed after thirty minutes. This change in dose consequence methodology for the subject event was made by the Palo Verde fuel vendor in 1993. The revised methodology was used to calculate the UFSAR section 15.3.4.3.1.C threshold fuel failure of 25 percent based on the current licensing basis 2-hour site boundary thyroid dose limit of 240 rem.

APS engineering evaluated the assumption change to close the stuck open ADV after 30 minutes under 10 CFR 50.59 and determined the methodology change would require NRC approval. As a result, the analysis was redone in accordance with the UFSAR methodology (i.e., without the stuck open ADV being manually closed). A review of reload analyses since 1993 was performed to ensure that the combination of predicted fuel failures and bounding radial peaking factor due to a reactor coolant pump shaft break would not have resulted in the 2-hour site boundary thyroid dose exceeding the 240 rem limit. For example, based on the methodology described in UFSAR section 15.3.4 and a dose limit of 240 rem, with a bounding radial peaking factor of 2.0, the maximum fuel failure would be approximately 15 percent. Likewise if the radial peaking factor were 1.4, then the maximum fuel failure would be approximately 21.5 percent. For each reload design, the 240 rem 2-hour site boundary thyroid dose was verified based on an assessment of calculated fuel failure and the corresponding radial peaking factor. Section D of enclosure 1 has been revised to describe the licensing basis methodology results. These results do not change the conclusions reached in the original submittal.

Large Steam Line Break Analysis

Subsequent to the submittal of the referenced letter, APS engineering personnel determined that the analysis for the large steam line break inside containment with a concurrent loss of offsite power used a nonconservative assumption. Specifically, work being done for the Unit 2 steam generator replacement and power uprate project identified that the analysis for the large steam line break inside containment with a concurrent loss of offsite power discussed in the referenced letter used the most negative moderator temperature coefficient (MTC). Since this large steam line break analysis assumes a simultaneous loss of offsite power, the event also becomes a loss of reactor coolant system (RCS) flow event. Since loss of RCS flow is a heatup event, the most positive or least negative MTC would result in the worst consequences.

In the evaluation of this incorrect assumption, APS engineering determined that the low reactor coolant flow trip was not required to mitigate a large steam line break event. In 1994, as part of the Palo Verde Reload Process Improvement Project (RPI), a large steam line break event that credited the low reactor coolant flow trip was evaluated due to reliability questions related to the environmental qualification of the RCP low shaft speed trip which normally would provide the protection for this event. This event assumed a simultaneous steam line break inside containment and loss of offsite power. This event was added to the safety analysis basis engineering documents, but was not added to the UFSAR since the large steam line break events described in UFSAR section 15.1.5 continued to be the bounding events

In light of the nonconservative MTC assumption and its affect on the analysis results, the reliability of the RCP low shaft speed trip was reevaluated. The evaluation determined that the RCP low shaft speed trip would reliably provide the protection function as required. This validated the original design basis and safety analysis basis for Palo Verde and negated the need to include a steam line break event that relied on the low reactor coolant flow trip in the design basis and the amendment request. Therefore, the reference to the large steam line break has been removed from the discussion supporting the proposed Technical Specification amendment in enclosure 1.

The problems discussed above were entered into and evaluated in accordance with the Palo Verde corrective action program. A review of other analyses is being performed in light of these issues.

Enclosure 1 has been revised to reflect the changes discussed above. Change bars have been added to identify the changes to enclosure 1 to the referenced letter. The technical specification pages submitted in reference 1 are not affected by these changes. Provided in enclosure 1 to this letter are the following sections which support the proposed Technical Specification amendments:

A. Need for the Amendment
B. Description of the Proposed Technical Specification Amendment
C. Purpose of the Technical Specification
D. Safety Analysis of the Proposed Technical Specification Amendment
E. No Significant Hazards Consideration Determination
F. Environmental Consideration
G. Revised Technical Specification Pages
H. Retyped Technical Specification Pages

In accordance with PVNGS Quality Assurance Program, the Plant Review Board and Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter this request is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

APS requests 60 days to implement the approved Technical Specification amendment. The 60 days is required to complete procedure changes, and complete and schedule modification packages for the setpoint changes in all three units.

The proposed amendment modifies the allowable values for the surveillance requirements associated with the steam generator low reactor coolant flow reactor protection system trips. Therefore, APS requests that the following condition be added to the amendment issuance letter: "For surveillance requirements associated with the revised allowable values for functions 12 and 13 in technical specification Table 3.3.1-1, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the date of implementation of this amendment." This is consistent with the license condition issued with technical specification amendment 117 to the Palo Verde operating license.

No commitments are being made to the NRC by this letter.

Should you have any questions, please contact Scott A. Bauer at (602) 393-5978.

Sincerely,

David Mauldin

CDM/SAB/RKR/mah

Enclosures

cc: E. W. Merschoff (all w/Enclosures)
M. B. Fields
J. H. Moorman
A. V. Godwin

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin
David Mauldin

Sworn To Before Me This 31st Day Of March, 2000.

Nora E. Meador
Notary Public

My Commission Expires

April 6, 2003

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/6/2003

ENCLOSURE 1

Proposed Amendment to Units 1, 2 and 3
Technical Specification 3.3.1

Proposed Amendment to Units 1, 2 and 3 Technical Specification 3.3.1

A. NEED FOR THE AMENDMENT

The Reactor Coolant Flow, Steam Generator #1-Low and Reactor Coolant Flow, Steam Generator #2-Low reactor protection system trips provide protection against a reactor coolant pump (RCP) sheared shaft event described in the UFSAR Chapter 15 "Accident Analysis." A reactor trip is initiated when the differential pressure across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint normally stays below the indicated differential pressure by a preset value called the Step function, unless limited by a preset maximum decreasing rate determined by the Ramp function, or by a preset minimum value called the Floor function. The Step function is the amount by which the trip setpoint remains below the input signal unless limited by Ramp or Floor functions. The Ramp function is the maximum permitted rate of decrease of the trip setpoint. There are no technical restrictions on the rate of increase of the trip setpoint. The Floor function is the enforced minimum value of the trip setpoint. The combined action of these functions (settings) determines the actual trip setpoint at any moment. The trip setpoint ensures that a reactor trip occurs to prevent violation of the peak linear heat rate (LHR) or departure from nucleate boiling ratio (DNBR) safety limits. There is a separate trip for each steam generator. Pre-trip alarms are also provided.

Since the variable trip setpoint will track the indicated differential pressure upwards very quickly, but is reduced very slowly, normal process noise will keep the setpoint much closer to the mean differential pressure signal than the Step function alone would indicate. This action is conservative with respect to the safety analysis assumptions, but it can result in a trip hazard depending on the magnitude of the noise. A large amount of noise on this process signal is to be expected since the signal is the difference of two pressures taken across a steam generator (a large and complex device) with the high flow rates that exist. Even if overall flow was constant, significant turbulence would still be expected where reactor coolant exits the steam generator.

In 1986 the PVNGS units experienced two plant trips caused by spurious operation of the low reactor coolant flow variable trip. The system vendor, Combustion Engineering determined that the steam generator differential pressure signal includes a random noise component. The source of the noise is believed to be related to the large fluid system acoustic waves propagating throughout the RCS, and randomly initiated by the natural turbulence of flow. The frequency character is a function of the fluid properties and the geometry of the system.

Technical Specification amendments 10 and 5 for Units 1 and 2, respectively, were subsequently issued. The Technical Specification amendments changed the variable trip setpoint (Step, Floor, and Ramp functions) so that process noise could be accommodated without tripping the units. Combustion Engineering also recommended that a small amount of additional filtering be added to the process instrumentation to eliminate spurious trips. Since the filter modification and Technical Specification amendments were implemented there have been no spurious full unit trips associated with the differential pressure signal.

However, since 1992, all three PVNGS units have experienced multiple-channel pre-trip alarms and/or single-channel trips which are attributed to the differential pressure signal. Palo Verde believes that this is a result of the random noise component discussed above. Recent investigation shows that the differential pressure signal periodically rises approximately three psid in six to eight seconds and then immediately drops by as much as six psid in about two seconds. During this sequence the variable setpoint will increase and then hold at the increased setpoint when the process signal drops back down. This often results in the average value of the process signal falling close to the setpoint. PVNGS data indicates that such pressure changes occur every 10 to 20 minutes. Depending on the magnitude of the pressure change, a pre-trip alarm or even a channel trip signal may occur. This process is seen in all three PVNGS units.

Although there is limited data, the frequency of these spurious pretrips appears to be increasing. This is attributed to the slowly increasing differential pressure across the steam generators over time, primarily due to steam generator tube plugging. As the differential pressure increases, the magnitude of the signal excursions due to the random noise component also increases. Therefore, as more steam generator tubes are plugged the potential for a spurious trip increases.

Palo Verde has determined that these excursions are not a result of hardware or instrumentation problems, but are fundamental to the system design. PVNGS Engineering has concluded that a change to the Technical Specification allowable values for the Ramp, Floor, and Step functions (i.e., lowering the effective setpoint) will directly increase the operating range and reduce the trip hazard associated with the random noise component. Additional filtering is not considered an option, since its effect on system response is relatively imprecise. Therefore, any changes to hardware or instrumentation are impractical.

B. DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The allowable values in Technical Specification section 3.3.1, Table 3.3.1-1, Item 12 "Reactor Coolant Flow, Steam Generator #1-Low" and Item 13 "Reactor Coolant Flow, Steam Generator #2-Low," will be changed from ≤0.118 psid/sec. to ≤0.115 psid/sec. for Ramp, from ≥11.7 psid to ≥12.49 psid for Floor, and from ≤10.2 psid to ≤17.2 psid for Step. This change is required to reduce the demonstrated spurious trip hazard associated with this setpoint.

C. PURPOSE OF THE TECHNICAL SPECIFICATION

The low reactor coolant flow trip function is part of the Reactor Protective System (RPS). The RPS initiates a reactor trip to protect against violating the core specified acceptable fuel design limits and breaching the reactor coolant pressure boundary (RCPB) during anticipated operational occurrences (AOOs). Specifically, the low reactor coolant flow trip function ensures that a reactor trip occurs to prevent violation of the peak LHR or DNBR safety limits. The protection and monitoring systems have been designed to ensure safe operation of the reactor.

D. SAFETY ANALYSIS OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The changes to the allowable values for the low reactor coolant flow trip function settings will provide a larger Step function between the process signal (indicated differential pressure) and the variable trip setpoint, while making the Floor and Ramp more restrictive. The overall effect of these changes will delay the RPS initiated low reactor coolant flow reactor trip. UFSAR Chapter 15 "Accident Analysis," identifies one event that relies on the low reactor coolant flow trip. This event involves a single RCP sheared shaft with a loss of offsite power (UFSAR 15.3.4). Therefore, the single RCP sheared shaft with a loss of offsite power event was reanalyzed to determine the effect of the delayed reactor trip on the analysis described in UFSAR Chapter 15.

In addition, other UFSAR events were evaluated to verify that these events were not affected by this change. The evaluation determined that the results of the bounding analyses for the other UFSAR events were not affected by this change.

RCP Sheared Shaft

The RCP sheared shaft event is a limiting fault event that results in a decrease in reactor coolant flow. Violation of the specified acceptable fuel design limits (SAFDLs) and resulting fuel failure is permissible. The dose consequences are the limiting factor for this event and are limited to the 10 CFR 100 limit (less than 300 Rem thyroid dose and 25 Rem whole body dose at the EAB). UFSAR Section 15.3.4.3 "Analysis of Effects and Consequences," currently states that "The resultant radiological consequences are a 2 hour site boundary thyroid dose of less than 240 Rem. This is within 10CFR100 guidelines."

For decreasing reactor coolant flow events, the major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a SAFDL has been violated and thus whether fuel damage could be anticipated. Those factors that cause a decrease in local DNBR are:

- increasing reactor coolant system (RCS) temperature,
- decreasing RCS pressure,
- increasing local heat flux (including radial and axial power distribution effects), and
- decreasing RCS flow.

During the first few seconds of the RCP sheared shaft transient, the combination of decreasing RCS flow and increasing RCS temperature results in a decrease in the fuel pins' DNBR. Minimum DNBR is reached at approximately 2 seconds when the RCS flow approaches the flow for three RCPs operating. The decrease in DNBR is reversed as a result of negative reactivity feedback via doppler and void coefficients. Following the reactor trip, a drop in power and heat flux results in rapid recovery of DNBR.

The doppler and void coefficients are primarily responsible for turning DNBR around once three RCP flow has been reached. The time of control rod insertion (i.e., timing of the reactor trip) primarily influences the rate of DNBR recovery and thus relates to DNBR propagation. These two distinct cause-and-effect relationships are fundamental to the sheared shaft event.

The reanalysis of the sheared shaft event using the same methodology as the original analysis, determined that the overall effect of the changes to the allowable values for the low reactor coolant flow trip function was to delay the RPS initiated low reactor coolant flow reactor trip for this event from the current value of approximately 1.2 seconds after event initiation to approximately 2.5 seconds after event initiation. The reanalysis of the sheared shaft event concluded that delaying the reactor trip would result in approximately the same minimum DNBR as previously analyzed. This is expected since the RCP coastdown characteristics are not being changed. Furthermore, although the time-in-DNB-condition (DNBR propagation) increases from approximately 2.6 seconds to approximately 3.9 seconds as a result of the delay in reactor trip, it remains below the limiting time (4.5 seconds) for the strain limit to be reached. Thus, DNBR propagation is also not a concern.

The reanalysis also evaluated the impact of extending the total trip time from approximately 1.2 seconds to approximately 2.5 seconds and assuming a LOP at approximately 3 seconds after the trip. The reanalysis showed that the minimum DNBR was relatively unchanged. This is expected because at 2 seconds into the event - close to the time of minimum DNBR - flow reaches the flow for three RCPs operating. Therefore, the reanalysis concluded that since minimum DNBR was relatively unchanged, the UFSAR section 15.3.4.3 EAB dose consequences of 240 Rem remains bounding for the sheared shaft event.

The methodology used for the reanalysis is consistent with the original methodology used in the CESSAR and UFSAR. A detailed description of the methodology is included in enclosure 2.

E. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an 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. A discussion of these standards as they relate to this amendment request follows:

Standard 1 -- Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed change will change the Reactor Protection System (RPS) reactor coolant flow trip setpoints. The RPS functions to mitigate the consequences of an accident. The changes to the low reactor coolant flow trip setpoints will reduce or eliminate unnecessary challenges to the RPS. Therefore, the proposed change will not involve a significant increase in the probability of an accident previously evaluated.

These changes will result in an increased time delay for the RPS low reactor coolant flow trip. The reanalysis of the affected Updated Final Safety Analysis Report (UFSAR) Chapter 15 event (UFSAR 15.3.4, Reactor Coolant Pump Shaft Break with Loss of Offsite Power), with the increased time delay, shows that the dose consequences for this event remains bounded by the UFSAR analysis. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

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

No. The proposed change will change the RPS reactor coolant flow trip setpoints. The RPS functions to mitigate the consequences of an accident. The changes to the low reactor coolant flow trip setpoints will reduce or eliminate unnecessary challenges to the RPS. The proposed change only changes the mitigating actions of the RPS, without changing the required function of the RPS. Therefore, the change to the low reactor coolant flow trip setpoints does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed change will change the RPS reactor coolant flow trip setpoints. The reanalysis of the affected UFSAR Chapter 15 event (UFSAR 15.3.4, Reactor Coolant Pump Shaft Break with Loss of Offsite Power), with the revised reactor coolant flow trip setpoints, shows that the minimum departure from nucleate boiling ratio (DNBR) and specified acceptable fuel design limits (SAFDLs) for this event remains bounded by the UFSAR analysis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the responses to these three criterion, APS has concluded that the proposed amendment involves no significant hazards consideration.

F. ENVIRONMENTAL CONSIDERATION

APS has determined that the proposed amendment involves no changes in the amount or type of effluent that may be released offsite, and results in no increase in individual or cumulative occupational radiation exposure. As described above, the proposed TS amendment involves no significant hazards consideration and, as such, meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

G. REVISED TECHNICAL SPECIFICATIONS PAGES

Units 1, 2, and 3: Page 3.3.1-9

H. RETYPED TECHNICAL SPECIFICATION PAGES

Units 1, 2, and 3: Page 3.3.1-9

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Steam Generator #1 Level - Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
9. Steam Generator #2 Level - Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
10. Steam Generator #1 Level - High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
11. Steam Generator #2 Level - High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
12. Reactor Coolant Flow, Steam Generator #1-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ ~~0.118~~ psid/sec. Floor: ≥ ~~11.7~~ psid Step: ≤ ~~10.2~~ psid
13. Reactor Coolant Flow, Steam Generator #2-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ ~~0.118~~ psid/sec. Floor: ≥ ~~11.7~~ psid Step: ≤ ~~10.2~~ psid

(continued)

Ramp: ≤ 0.115 psid/sec.
Floor: ≥ 12.49 psid
Step: ≤ 17.2 psid

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
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9. Steam Generator #2 Level – Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≥ 43.7%
10. Steam Generator #1 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
11. Steam Generator #2 Level – High	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	≤ 91.5%
12. Reactor Coolant Flow, Steam Generator #1-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid
13. Reactor Coolant Flow, Steam Generator #2-Low	1,2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13	Ramp: ≤ 0.115 psid/sec. Floor: ≥ 12.49 psid Step: ≤ 17.2 psid

(continued)

ENCLOSURE 2

Requested Information from August 17, 1999
and October 5, 1999 Phone Conversations

The PVNGS reload analysis methodology, including an overview of the seized rotor/sheared shaft analysis, was reviewed and approved by NRC in the PVNGS Reload Analysis Methodology Report (dated June 14, 1993). The sheared shaft methodology for this Technical Specification (T. S.) submittal is based on the approved methods described in the UFSAR (Refer to 1.6, "Material Incorporated by Reference," 4.3.3, "Analytical Methods," 4.3.4, "References," 4.4.7, "References," and 15.3.4, "Reactor Coolant Pump Shaft Break with Loss of Offsite Power."), but includes a revision to selected input assumptions. This description is provided to clarify previously approved methods and inputs.

I. Sheared Shaft/Seized Rotor (SS/SR) Analysis Methodology

Beginning with Unit 2 Cycle 7, a limited long-term scenario (50 seconds duration) was performed to evaluate the impact of stretch power on the sheared shaft analysis.

Subsequently, in Unit 1 Cycle 7, a short-term scenario for the SS/SR event was performed as part of Reload Process Improvement (RPI) in order to develop a bounding analysis for evaluating fuel failure. The following text details the methodology employed in the bounding RPI analysis, and in the analysis for the requested change to the Technical Specifications.

Initial Power Operating Limit (POL) Calculation:

The selection of initial conditions is based upon the criterion of preserving as much subcooling as possible while maintaining reasonable power operating limit (POL) conditions from the operational standpoint. Preserving initial subcooling minimizes negative reactivity feedback due to voiding associated with loss of flow through the core. Hence, initializing the transient from these conditions maximizes predicted fuel failure. The thermal-hydraulics code CETOP is utilized to calculate the POL conditions. The Core Operating Limit Supervisory System (COLSS) is utilized to preserve initial margin. The calculated POL conditions are listed in Table A-1 to illustrate the selection of initial conditions. Section III details the selection of initial plant conditions.

MTC Tuning:

HERMITE[1] is utilized to perform moderator temperature coefficient (MTC) calculations and determine the soluble boron concentration that corresponds to the limiting MTC value (See Section III).

1. CENPD-188, "HERMITE, a Multi-Dimensional Space-time Kinetics Code for PWR Transients," March 1976 (Proprietary).

Transient Simulations:

Initiating from the POL conditions and boron concentrations noted above, the HERMITE/GENI/CETOP codes are utilized to simulate the transient response of the Sheared Shaft and Seized Rotor events. The main difference between these two transients is the Reactor Coolant Pump (RCP) coastdown curve (independently generated by the COAST code) and the credited Reactor Protection System (RPS) response (e.g. reactor trip and associated delays).

For the Seized Rotor event, a Core Protection Calculator System (CPCS) RCP Shaft Speed trip is credited. The CPCS will generate a signal to the trip breakers 0.71 seconds after the RCP seizes. The trip breakers response time is set at 0.15 seconds. Thus, the trip breakers are credited to open at 0.86 seconds.

For the Sheared Shaft event, the Steam Generator (SG) Low RCS Flow trip, also referred to as the "SG dP Low RCS Flow Trip", is credited to occur at 90% loop flow (corresponds to ~95% RCS flow). The trip is based on differential pressure across the SG primary side as the measurement input[1]. The total response time, including the trip breakers response time of 0.15 seconds, is 1.20 seconds (See Table A-3).

The "SG dP Low RCS Flow Trip" was analyzed with a total trip time changed from 1.20 seconds to 2.50 seconds in order to support the proposed Plant Protection System (PPS) setpoint for the "SG dP Low RCS Flow" reactor trip. The "SG dP Low RCS Flow Trip" has been responsible for a long series of spurious pre-trips and trip signals. Because significant extra time was found in the safety analysis, the option of a Technical Specification change was evaluated which resulted in a RPS total trip time of 2.50 secs, rather than previously credited 1.20 secs (see Tables A-3 and A-4). The reanalysis of the Sheared Shaft Event showed that the results (i.e., the minimum DNBR and transient time in DNB) are in close agreement and acceptable with those established in the RPI Analysis of Record (AOR).

The purpose of the HERMITE/GENI/CETOP transient simulations is to determine plant parameters that correspond to the time of minimum DNBR. Table A-2 lists the bounding time-of-minimum DNBR conditions resulting from the RPI SS/SR simulations. Note that these conditions were based on the Sheared Shaft event, which proved to be slightly more adverse than the Seized Rotor.

Table A-3 lists a typical Sequence of Events for the Short-Term Sheared Shaft event for the RPI AOR and Table A-4 lists a typical sequence of events for the Short-Term Sheared

1. A PVNGS Design I&C calculation translates the T.S. values and hardware settings to support this assumption.

Shaft event for the proposed T. S. change. Note that the transient simulations do not include the Loss of Offsite Power.

Thermal Hydraulic Calculation:

The core thermal-hydraulics code TORC[1] was utilized to calculate DNBR values at various integrated radial peaking factors (Fr), and to account for the effects of the 3-pump flow and time of minimum DNBR conditions.

The result of these 3-Pump TORC cases were a set of Fr versus DNBR data which was subsequently used to calculate fuel failure. Table A-5 lists a set of "typical" data resulting from the 3-Pump TORC cases.

Fuel Failure Calculation:

The calculated fuel failure associated with the reload pin census is based on the Fr versus DNBR data using statistical convolution. Typical values range from 5% to 16%.

In future reload cycles, the fuel failure will be calculated based upon the cycle specific pin census, DNBR statistics, and the Fr versus DNBR data generated in the bounding analysis (Table A-5).

Assessment of DNBR Propagation:

Under severe local conditions, channel blockage due to fuel rod ballooning may potentially impact the heat transfer of adjacent rods sufficiently to produce DNB Propagation. A mechanistic evaluation of fuel clad ballooning in fuel rods experiencing severely degraded heat transfer (i.e. DNB) with internal pressure exceeding RCS pressure was documented. The objective was to provide a means of evaluating the potential of DNB Propagation in accordance with the NRC approved Topical Report CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

This methodology was utilized to evaluate the fuel clad strain for a wide range of local conditions. The principle result of the evaluation was a determination of the minimum time in DNB which would allow fuel rods to reach the NRC approved maximum strain limit.

In accordance with the general procedure for applying the DNB propagation methodology to Non-LOCA transients, the local conditions were extracted from the HERMITE/GENI/CETOP transient simulation. Since this simulation provides the dynamic thermal hydraulic response to the transient, it was chosen over the detailed, static TORC DNBR calculation (which analyzed the time of minimum DNBR only). Note that the CETOP

1. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April, 1986.

calculated DNBR values utilized in the analysis originate from the 4-Pump CETOP Model, not the 3-Pump model. However, local conditions from the 4-Pump model are valid for determining DNB propagation provided the appropriate hot channel flow factor is applied to the core average mass flux.

For the proposed T.S. change, the duration of time in DNB (DNBR < 1.30) experienced by the SS/SR event is less than the minimum time required to reach the NRC approved strain limit. Therefore, the fuel clad will not balloon enough to impede the channel flow sufficiently to propagate DNB to adjacent fuel rods.

Table A-1 Typical Initial POL Conditions

Parameter	Selection Strategy	Input to 4-Pump SAFDL Calculation	4-Pump POL Conditions
Core Average Heat Flux (E6 Btu/hr-ft2)	Maximize	0.224805 (119% ROPM)	0.188912
Core Mass Flux (E6 lbm/hr-ft2)	Maximize	3.0015	
Inlet Temperature (F)	Minimize	548	
RCS Pressure (psia)	Maximize	2242	
Fr[a]	Maximize	2.00	
Axial Power Distribution (ASI)	Maximum Bottom Peaked	+0.19535	
Calculated DNBR	- -	1.300	1.6507

a. A value of 1.70 was used in the dose calculation. This does not impact the POL calculation.

Table A-2 Typical Sheared Shaft Transient Response at Time of Minimum DNBR Condition

Parameter	Initial 4-Pump POL Conditions	Typical Change	RPI Bounding Change	Final Conditions for 3-Pump TORC
Core Average Heat Flux (E6 Btu/hr-ft2)	0.188912	0.976	1.00	0.188912
Core Mass Flux (E6 lbm/hr-ft2)	3.0015	0.746	0.70	2.10105
Inlet Temperature (F)	548	- -	- -	548
RCS Pressure (psia)	2242	- -	- -	2242
Fr	2.00	+0.025	+0.05	2.05
Axial Power Distribution (ASI)	+0.195	+0.002	+0.002	+0.197

Table A-3 Typical Sheared Shaft Short-Term Sequence of Events

(Loss of Offsite Power Not Included in Short-Term scenario)

Time (seconds)	Transient	RPS Response	CETOP DNBR
0.0	Initiate Sheared Shaft Event - POL Conditions		1.650
0.2	RCS Flow ~ 95.4%	Low RCS Flow Trip Set-point Reached (S.G. ΔP)	1.573
0.90	RCS Flow ~ 81.9%	RPS Delays	1.30
1.20	RCS Flow ~ 78.5%	Trip Signal Generated	1.278
1.54	RCS Flow ~ 75.7%	CEDM Hold Coil Delay	1.230
2.00	RCS Flow ~ 74.6%	CEAs Inserted ~ 6.9%	1.210 Minimum DNBR
3.00	RCS Flow ~ 74.3%	CEAs Inserted ~ 33.2%	1.251
3.48	RCS Flow ~ 74.3%	CEAs Inserted ~ 48.0%	1.30
4.00	RCS Flow ~ 74.3%	CEAs Inserted ~ 64.1%	1.502
5.00	RCS Flow ~ 74.3%	CEAs Inserted ~ 86.7%	1.782

Table A-4 Typical Sheared Shaft Short-Term Sequence of Events with Increased RPS and CEDMCS Holding Coil Delay Time
(Loss of Offsite Power Not Included in Short-Term scenario)

Time (seconds)	Transient	RPS Response	CETOP DNBR
0.00	Initiate SS Event	None	1.650
0.90	RCS Flow ~ 81.9%	Fuel begins DNB Condition	~1.34
2.00	RCS Flow ~ 74.6%	RPS Delays, Minimum DNBR for the transient	1.209 Minimum DNBR
2.50	RCS Flow ~74.6%	Trip Signal Generated	1.211
3.10	RCS Flow ~ 74.3%	CEDMCS HC Delay, Rods begin to insert	1.212
3.50	RCS Flow ~ 74.3%	CEAs Inserted ~5%	1.214
5.50	RCS Flow ~ 72.2%	CEAs Inserted ~70%	1.520
6.00	RCS Flow 69.3%	CEAs Inserted ~80%	1.598
7.00	RCS Flow ~ 63.8%	CEAs Inserted ~ 100%	1.720
10.00	RCS Flow ~ 59.0%	None	2.242

Table A-5 Typical Fr versus DNBR Data

Integrated Radial Peak	Minimum DNBR
2.10	0.606
2.00	0.891
1.90	1.128
1.80	1.306
1.70	1.442
1.60	1.590

II. Historical Background

PVNGS Cycle 5

Following the PVNGS-2 Cycle 5 Reload, an expert team review was performed as part of the validation of the reload team concept. This detailed review concluded that the selection of initial POL conditions should be reevaluated. Specifically, the team determined that selection of a minimal RCS pressure (along with a minimal integrated radial peak) results in voiding in the upper region during the transient which introduces negative reactivity. To minimize this negative void reactivity, the selection of initial conditions was changed to preserve as much subcooling as possible (refer to typical values in Section III of this attachment and to Table A-6).

One result of the change in selection of initial conditions was that the Seized Rotor event was no longer bounding relative to the Sheared Shaft event. The Sheared Shaft yielded slightly higher fuel failure. This revised methodology was first applied to the PVNGS-1 Cycle 5 Reload. The calculated fuel failure for this reload, 5.76%, exceeded the previously reported UFSAR value of 4.5%. A new SS/SR 2-hour thyroid dose calculation was performed to demonstrate that the previously reported UFSAR 2-hour thyroid dose of 240 REM was not exceeded. The calculation documented a 2-hour thyroid dose of less than 200 REM, based on 12% fuel failure.

The improved inlet flow distribution, a new DNBR SAFDL (1.30 versus 1.24), and DNB statistics were incorporated into the PVNGS-3 Cycle 5 Reload. A 3-Pump TORC Model based upon the improved inlet flow distribution was also developed. The result of incorporating these improvements was a significant decrease in calculated fuel failure. The PVNGS-3 Cycle 5 SS/SR Analysis calculated less than 0.85% fuel failure.

PVNGS Cycle 6

Following PVNGS-3 Cycle 5, the Cycle 6 Reloads calculated fuel failures less than 1.0%. The PVNGS-3 Cycle 6 Reload adopted the "No Clad Lift-Off" topical and was required to demonstrate that no DNB Propagation would occur. Future SS/SR analyses would also need to demonstrate no DNB Propagation.

PVNGS Cycle 7

Stretch Power and the associated plant changes (i.e. increased tube plugging, inlet temperature LCO, etc.) only slightly impacted the PVNGS-2 Cycle 7 SS/SR Analysis. For this reload (U2C7), both the short-term and long-term scenarios were analyzed. A revised UFSAR write-up was submitted which documented less than 0.2% fuel failure (hence, less than 240 REM 2-hour site boundary thyroid dose). For RPI (beginning with Unit 1 Cycle 7), a revised source term was also developed (see Section IV).

Post Cycle 7 Reloads

Reload Process Improvement attempted to bound the transient and thermal-hydraulic response of the SS/SR events for future reloads. Specifically, a cycle-independent set of Fr versus DNBR data based upon "bounding" physics and plant data are verified for each reload during the reload analysis process. This bounding set of Fr versus DNBR data is then used, along with the cycle specific reload pin census file, to calculate fuel failure. Reload fuel failure is then compared against the "bounding" fuel failure which yields the reported dose limits. If necessary, a cycle specific analysis is performed.

Table A-6 PVNGS Sheared Shaft Transient-Selection of Initial Conditions[a]

Case	CAHF	Mass Flux	T-Inlet	Pressure	Fr	AXPD	Transient ROPM
Limiting Axial Power Distribution							
1	Max-Iterated	Maximum	Minimum	Minimum	Minimum	Top	1.2226
2						Bottom	1.2325
Limiting RCS Pressure							
3	Max-Iterated	Minimum	Minimum	Minimum	Minimum	Bottom	1.206
4							1.237
Limiting Core Mass Flux							
5	Max-Iterated	Minimum	Minimum	Maximum	Minimum	Bottom	1.237
6		Maximum					1.2552
Limiting Radial Peak							
A maximum radial peaking factor allows for the use of maximum pressure, minimum temperature, and maximum mass flux when determining the POL conditions. This combination of initial condition minimizes the void reactivity feedback effects and yields lower transient DNBR values. Note that no credit is taken of a 'peakier' pin census file (the pin census associated with higher radial peaks, like those associated with rodded operation, would yield beneficial affects on calculated fuel failure)'							
Original U2C5 Seized Rotor	Maximum	Minimum	Minimum	Minimum	Minimum	Top	1.194 (2.16% fuel failure)
Revised U2C5 Seized Rotor	Maximum	Maximum	Minimum	Maximum	Maximum	Bottom	1.245 (9.28% fuel failure)

a. Blank cells indicate that there is no change in the equivalent parameter between comparative cases.

III. Plant Initial and Other Event Dependent Conditions

Inputs to the SS/SR analysis should be based upon the criterion of preserving as much subcooling as possible while maintaining reasonable POL conditions from the operational standpoint. The "typical" values listed below were obtained from the Bounding RPI SS/SR Analysis.

Licensed Power Limit (LPL):

A maximum rated core power (i.e. 100% T.S. LPL) increases the consequences of the SS/SR events. Since the event is initiated from a POL, power measurement uncertainties (i.e. 2% secondary calorimetric uncertainty) need not be added to the initial power level. Typically, the initial power level is set to 3876 MWt. Note - while not explicitly included in the transient, a 2% power measurement uncertainty is accounted for in the source term used to determine dose.

Core Average Heat Flux (CAHF):

A maximum CAHF increases the consequences of the SS/SR events. CAHF is calculated based upon the power level and number of fuel pins. Due to the displacement of B_4C shims by Erbium fuel rods, the number of fuel rods have been increasing (until all erbium fuel management). This growing trend in the number of fuel rods reduces the calculated CAHF. Typically, a smaller than expected number of fuel rods would be used to calculate CAHF. This quantity of fuel rods would be calculated as: 241 assemblies x 236 pins - 752 non-fuel pins = 56,124 fuel pins.

Primary Coolant Flow:

A maximum primary coolant flow preserves the largest subcooling in the upper region of the core which minimizes negative void reactivity feedback during the transient. Typically, a maximum coolant flow of 115% of design (maximum RCP capacity) would be utilized. This corresponds to a core mass flux of 3.0015 E6 lbm/hr-ft^2 (115% of 2.61 E6 lbm/hr-ft2) and a core mass flow of 182.9 E6 lbm/hr (115% of (164 E6 lbm/hr - 3% core bypass)).

Inlet Temperature:

A minimal inlet temperature preserves the largest subcooling in the upper region of the core which minimizes negative void reactivity feedback during the transient. The inlet temperature is set to the lower LCO minus "monitoring" uncertainty. Typically, the inlet temperature is set to 548 °F (LCO 550 °F - 2.0 °F monitoring uncertainty).

RCS Pressure:

A maximum RCS pressure preserves the largest subcooling in the upper region of the core which minimizes negative void reactivity feedback during the transient. The proper selection of other initial parameters (especially a large integrated radial peak) will ensure a maximum RCS pressure. An iterated pressure between 2025 - 2300 psia was used in the bounding analysis.

Integrated Radial Peak (Fr):

A maximum radial peaking factor preserves the largest subcooling in the upper region of the core, which minimizes negative void reactivity feedback during the transient. Hence, the initial POL conditions are governed by the hot channel which is far different from the average channel. The radial peaking factor is set to the cycle maximum value predicted for the reload. This allows the use of a maximum RCS pressure in the initial POL conditions. Typically, a radial peaking factor of 2.0 is utilized. A value of 1.70 was used in the dose calculation. This does not impact the POL calculation.

Axial Power Distribution:

A bottom peaked axial power distribution delays the power decrease due to scram. In addition, generation of a majority of the power in the region where the core is most subcooled (bottom of the core) reduces the negative void reactivity feedback. The axial power distribution is set to the "limiting" axial shape within the Analysis Range. The Analysis Range is the COLSS LCO ± uncertainty. A typical analysis value is +0.20 ASI at full power.

Moderator Temperature Coefficient:

A more positive (less negative) MTC increases the positive reactivity insertion due to moderator temperature feedback during the flow coastdown. The MTC is set to the most positive value allowed by the COLR. Typically, an MTC value of 0.0 E-4 $\Delta\rho/^{\circ}F$ is utilized.

Fuel Temperature Coefficient:

A more positive (less negative) FTC decreases the negative reactivity insertion due to fuel temperature feedback. The FTCs are tuned to bounding values in the HERMITE Models and verified during the reload analysis process.

Kinetics:

A maximum Beta fraction (β) delays the core power decrease after reactor trip which results in a later DNBR turn-around and a lower flow at the time of minimum DNBR. The kinetics parameters (β, λ, l*) are tuned to bounding values in the HERMITE Models and verified during the reload analysis process.

Net Scram Worth:

A minimum scram worth delays the core power decrease after reactor trip which results in a later DNBR turn-around and a lower flow at the time of minimum DNBR. The net scram worth (with Worst Rod Stuck Out) calculated based on the T.S. Power Dependent Insertion Limits (PDILs), is tuned to the bounding value in the HERMITE Models and verified during the reload analysis process. Typically, a minimum net scram worth of -7.0%$\Delta\rho$ is utilized.

Fuel Pellet/Clad Gap Conductance (Hgap):

A minimum gap conductance (Hgap) delays the core heat flux decrease after reactor trip which results in a later DNBR turn-around and a lower flow at the time of minimum DNBR. The minimum Hgap values are verified during the reload analysis process.

Flow Coastdown:

A maximum coastdown (1 RCP seized rotor/sheared shaft) minimizes the RCS flow at the time of minimum DNBR. The flow coastdown values are verified during the reload analysis process. Typical coastdowns for the SS/SR events are listed in Table A-7.

Table A-7 Typical RCS Flow Coastdown

Time (seconds)	Sheared Shaft (fraction)	Seized Rotor (fraction)
0.0	1.0	1.0
0.5	0.885	0.839
1.0	0.803	0.773
1.5	0.759	0.753
2.0	0.746	0.749
9.0	0.743	0.748

Reactor Protection System (RPS) Response:

For the Seized Rotor event, a Core Protection Calculator System (CPCS) RCP Shaft Speed trip is credited as soon as the RCP seizes. The CPCS will generate a signal to the trip breakers in 0.71 second (trip breakers response time equals 0.15 second). Thus, the trip breakers are credited to open at 0.86 second.

For the Sheared Shaft event, an "SG dP Low RCS Flow Trip" is credited at 90% loop flow (corresponds to ~95% RCS flow). The trip is based on differential pressure across the SG primary side as the measurement input. The total response time, including the trip breakers response time of 0.15 second, is 1.20 seconds (See Table A-3).

For this submittal, the "SG dP Low RCS Flow Trip" was reanalyzed with the total trip time changed from 1.20 seconds to 2.50 seconds in order to support the proposed Plant Protection System (PPS) setpoint for the "SG dP Low RCS Flow Trip." The "SG dP Low RCS Flow Trip" function has been responsible for a long series of spurious pre-trips and trip signals. Significant extra time was found in the safety analysis space, therefore, the option of a Technical Specification change was evaluated which resulted in a RPS total trip time of 2.50 seconds rather than the previously credited 1.20 seconds (see Tables A-3 and A-4). The reanalysis of the sheared shaft event showed that the fuel failure results are acceptable with respect to those established in the RPI AOR.

CEA Scram Curve:

An extended CEA drop time (scram position versus time) delays the core power decrease after reactor trip which results in a later DNBR turn-around and a lower flow at the time of minimum DNBR. The CEA scram curve is verified during the reload analysis process. A typical scram curve is listed in Table A-8.

Table A-8 Typical Scram Curve

Time (seconds)	Scram Insertion (%)
0.0	0
0.6	0
1.0	5
1.01	10
1.39	20
1.70	30
2.02	40
2.34	50
2.66	60
3.00	70
3.40	80
4.00	90
4.78	100

DNBR Probability Statistics:

The DNBR Probability Statistics are utilized in the fuel failure calculation using approved statistical convolution methods. Typical values used in the convolution technique are listed below.

- DNBR SAFDL[1] = 1.34
- Mean = 1.0605

1. This value has varied during recent reload analyses and its impact on fuel protection was incorporated into the cycle specific CPC addressable constants while leaving the DNBR SAFDL at 1.30.

IV. Dose Calculations for SS/SR Event

A revised dose calculation was generated as part of the Reload Process Improvement program. The calculation was notable in providing an alternate methodology to integrate the activity release based on activity transport and an emergency plant cooldown profile (100 F°/hour) for the unaffected steam generator. The activity release rate was modeled with the equation:

$$\frac{d}{dt}A = i(t) - K \cdot A(t)$$

where A is the activity in the steam generator at time t, i(t) is the rate at which Iodine enters the steam generator, and K is the effective steaming rate including partitioning. The solution is:

$$A(t) = e^{-K \cdot t}\left(\int i(t) \cdot e^{K \cdot t} dt + C\right)$$

and the activity release at any time t is:

$$Release = \int \frac{K \cdot A(t)}{M(t)} dt$$

where M(t) is the steam generator water mass at any time t.

Other inputs and assumptions are summarized below:

- The primary-to-secondary leakage rate is 720 gpd per SG, or 0.5 gpm per SG. These values are based on the Technical Specifications which were in effect at the time the RPI analysis was performed.[1]
- The initial activity concentrations are based on the Technical Specification limits on primary and secondary activity (dose equivalent iodine-131 limits of 1.0 μCi/gm primary and 0.1 μCi/gm secondary).
- The steaming rate is based on the steaming required for plant cooldown, which in turn is dependent on reactor decay heat (including actinides) and stored energy. The activity source term was based on the revised source term developed for RPI, beginning with Unit 1 Cycle 7.
- Offsite power is not available.
- An ADV is assumed to stick open from 1800 seconds for the duration of the event for the 2 hour EAB thyroid dose calculation.
- A partitioning factor of 100 for bulk boiling was used whenever a steam/water interface exists. A plant cooldown rate of 100 F°/hr was used which is more adverse than adminis-

1. These limits bound the current T.S. 3.4.14 limit of 150 gpd for total primary-to-secondary leakage, and 720 gpd is deemed to remain valid for this event.

trative control limit of 75 F°/hr. The RSB SER - CESSAR SYSTEM 80 documents acceptance of the use of an iodine partition factor of 100 for use in the SS/SR event for releases from the SG without the stuck open ADV.[1]

- Iodine is assumed to be released to the atmosphere with a partition coefficient of 1.0 for fluid leaked from primary to the SG with a stuck open ADV.
- Parameters and values used to evaluate dose consequences include:

  a. Power level - 102% Rated Thermal Power.

  b. Source Term developed for U2C7 Stretch Power submittal.

  c. Percent of fuel (bounding value) assumed to experience DNB - 17%.

  d. A bounding radial peaking factor (Fr) of 1.70

  e. RCS activity before event - 1.0 μCi/gm

  f. Secondary system activity before event - 0.1 μCi/gm

  g. ICRP 30 dose conversion factors as stated in PVNGS Technical Specifications.

  h. Atmospheric dispersion factors, X/Q, based on 1986-1991 site specific meteorological data.

The threshhold used for reload evaluation has been set at bounding value of 17% fuel failure, based on the assumptions noted above. This value results in an 2 hour EAB thyroid dose of 240 REM, which is less than the 10CFR Part 100 requirements.

V. Conservatisms

The following conservatisms exist in the RPI SS/SR bounding analysis.

a. The RPI bounding analysis assumed a transient change in core mass flux of 0.70 (fractional) when determining the time of minimum DNBR conditions. This conservative value (relative to the actual change of 0.745) may be used to justify future changes in the flow coastdown or future changes in the RPS setpoints and/or response times.

b. The RPI bounding analysis does not credit the core average heat flux decrease during the transient when determining the time of minimum DNBR conditions. This conservative assumption (relative to the actual 2% decrease) may be used to justify that future changes in physics (i.e. FTCs, MTC, kinetics, etc.) and/or fuel performance (i.e. Hgap) parameters will not affect the results of the bounding analysis.

c. The RCP coastdown curves are based on 90% pump inertia.

1. Docket No. 50-470, October 17, 1981.