Attachment 2

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

AND

CORRECTION OF SBLOCA ANALYSIS ERROR

(NON-PROPRIETARY)

Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna and Surry Power Stations Units 1 and 2

<u>Response to NRC Request for Additional Information (Non-proprietary)</u>

RAI 1

<u>Request:</u>

The proposed additions to the TS Core Operating Limits Report (COLR) References for both NAPS and SPS include reference to LTR EMF-2328(P)(A), and supplemental licensing reports ANP-3467P and ANP-3676P for NAPS and SPS, respectively. The licensing reports document plant-specific aspects of EMF-2328(P)(A), the implementation of a fuel vendor-independent modeling approach, and an additional modeling change. The formatting guidance provided in NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," indicates that plant-specific methodology should include a reference to the NRC staff approving safety evaluation. In addition, guidance contained in NRC-endorsed document Nuclear Energy Institute (NEI)-96-07, "Guidelines for 10 CFR 50.59 Implementation," indicates that applicable terms, limitations, and conditions of the use of a method of evaluation are documented not only in a methodology report such as ANP-3467P or ANP-3676P, but also in related documentation including the NRC staff SE approving such use. The inclusion of the NRC staff approving SE in the TS COLR References is reflective of the guidance in NEI 96-07. Please provide a justification describing how the applicable terms, limitations and conditions of the use of the plant-specific variants to EMF-2328(P)(A) will be reflected in the plant licensing bases, given that the staff approving SE is not proposed for referencing in the NAPS and SPS TS. Alternatively, provide revised TS COLR References revisions that conform to the guidance in GL 88-16.

<u>Response:</u>

Consistent with the guidance provided in GL 88-16 and NEI 96-07 for referencing plant-specific evaluation methodologies, the proposed Technical Specifications (TS) COLR references in Surry Power Station (SPS) TS 6.2.C (Reference No. 3) and North Anna Power Station (NAPS) TS 5.6.5.b.3 are revised to include the associated NRC Safety Evaluation Report as follows:

Surry TS:

3. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," as supplemented by ANP-3676P, "Surry Fuel-Vendor Independent Small Break LOCA Analysis," as approved by NRC Safety Evaluation Report dated [DATE].

North Anna TS:

3. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," as supplemented by ANP-3467P, "North Anna Fuel-Vendor Independent Small Break LOCA Analysis," as approved by NRC Safety Evaluation Report dated [DATE].

Proposed marked-up and typed TS pages are provided in Attachments 3 and 4, respectively.

RAI 2

Request:

Section 2.1 of Supplement 1 (P)(A) to EMF-2328(P)(A), Revision 0, describes that the methodology was intended to cover a range of breaks up to, and including, 10% of the cold leg cross-sectional area, at which point the break spectrum will be covered by the large break (LB) LOCA EM, EMF-2103(P)(A), Revision 0. However, NAPS and SPS currently have a LBLOCA EM that is not EMF-2103(P)(A), Revision 0, in their licensing bases. To ensure limiting LOCAs are analyzed to adequately demonstrate compliance with 10 CFR 50.46, please discuss any portions of the postulated LOCA break spectrum that would not be covered by the FVI-SBLOCA methodology and current licensing basis LBLOCA EMs in the plants' licensing bases. If there are breaks that are not considered, please provide an analysis of these breaks or provide justification that no explicit analysis of these breaks is necessary. Please discuss how Dominion would address similar situations in the future, in the event that different LBLOCA EMs are adopted with different analysis domains.

Response:

As identified in the FVI-SBLOCA License Amendment Request (LAR) for both stations, the analyzed break spectrum of the FVI-SBLOCA analysis covers a range of 1.0 inch to 8.7 inches in diameter. The current licensed LBLOCA analyses with ASTRUM define the LBLOCA spectrum as pipe breaks between 1 ft^2 area (approximately 13.5 inches in diameter) and two times the area of the cold leg.

The gap between large break and small break spectrums has been demonstrated to be a non-limiting region for the LOCA event. As evidenced by the FVI-SBLOCA break spectrum results for larger small breaks, sufficient coolant inventory exists for break flow to remove the stored energy from the core. Also, since depressurization occurs much faster for these breaks, low head safety injection initiates much sooner, mitigating progression of the event. Similarly, results from the existing ASTRUM LBLOCA analysis indicate that smaller (split) breaks in the spectrum are nonlimiting.

Thus, the postulated break spectrum considered in the ASTRUM LBLOCA analysis and the FVI-SBLOCA analysis ensure the limiting LOCAS are analyzed, demonstrating the requirements of 10 CFR 50.46 are met.

In the event that a different LBLOCA EM is adopted with different analysis domains, it would be verified that the combination of the SBLOCA and LBLOCA analyses continue to cover the full range of break sizes or demonstrate that the uncovered region remains non-limiting.

RAI 3

Request:

For the spectrum of analyzed break sizes, provide a description of the modeling of the break flow with regard to liquid/vapor composition. In practice, break orientation would, to an extent, dictate the predominance of liquid or vapor in the flow out of the break. Please justify how the modeling approach used considers the range of potential conditions associated with varied break locations, and how the modeling approach is limiting, or that the composition of the break flow effluent is insignificant when predicting an upper bound relative to the 10 CFR 50.46(b) acceptance criteria.

Response:

Framatome SBLOCA EM break flow modeling is based on the requirements of 10 CFR 50 Appendix K. The Moody correlation is used to determine mass flux at the break. Implementation of the model into the EM dates back to Siemens RELAP5 ANF models. Moody critical flow rate predictions have been, historically, shown to be conservatively high. Moody is a homogenous flow correlation and generates flows predicated on source node stagnation enthalpy and stagnation pressure. As a result, orientation is not a factor in the use of this correlation. A wide set of variations in event timing (for example break uncovery) are examined in the EM application via the range of breaks (1" - 10% of the cold leg area) analyzed. [

] This provides confidence that the most limiting set of break flow simulations are considered in the EM application to ensure that an upper bound relative to the 10 CFR 50.46(b) acceptance criteria has been realized.

RAI 4

Request:

EMF-2328, Revision 0, Supplement 1 (P)(A) makes note of several changes to the plant nodalization scheme and to parameters within the EM that should be utilized when performing SBLOCA analyses. ANP-3676P and ANP-3467P address the majority of these methodology changes. Please provide information to verify the following modeling approaches have been performed consistent with the methodology described in EMF-2328(P)(A), Revision 0, Supplement 1 (P)(A):

a. [ ]

b. [ ]

c. [ ]

d. The maximum positive moderator temperature coefficient allowable by TS is incorporated in the analyses, and

e. Fuel temperature reactivity feedback or other negative feedback mechanisms (e.g., void) are modeled using minimum calculated values.

Response:

a. [ ] This is illustrated in the summary report noding diagrams (for the North Anna summary report, ANP-3467P - see Figure 3-1 and the Surry summary report, ANP-3676P - see Figure 3-1). S-RELAP5 base deck preparation documentation and related inputs were reviewed for assurance. [ ]

b. [ ] The nodal diagrams for the reactor vessel S- RELAP5 model is included in the summary reports (North Anna document ANP-3467P - see Figure 3-3 and Surry document, ANP-3676P - see Figure 3-3). S-RELAP5 base deck preparation and model initialization documents as well as model inputs were reviewed for confirmation. [ ]

c. [

] This is illustrated in the summary report noding diagrams (for North Anna summary report, ANP-3467P - see Figure 3-1 and Surry summary report, ANP-3676P - see Figure 3-1). S-RELAP5 base deck preparation documentation and related inputs were reviewed for assurance.

d & e. The reactivity feedback inputs supplied to Framatome conform to the descriptions in EMF-2328(P)(A), Revision 0, Supplement 1 (P)(A). Specifically, the modeling approach described in sections 4.2, 4.3, and 4.4 of EMF-2328(P)(A), Supplement 1 (P)(A) was used.

Reactivity feedback corresponding to moderator density changes was provided as input to Framatome. The simulated reactivity defects were biased to be representative of a core with a BOC HFP MTC at the Technical Specification limit.

The inputs supplied to Framatome contain the most positive (least negative) reactivity feedback achievable as the moderator density decreases, which ensures that the negative reactivity feedback is minimized as the core voids.

The calculations were performed at ARO conditions. This produced a more positive MTC than would be achieved by modeling the control rod insertion, and resulted in less negative reactivity feedback as the core voids.

Last, the conversion of reactivity from pcm to dollars is accomplished by utilizing a large value of beta-effective.

RAI 5

Request:

Section 4.3.1 of ANP-3467P for NAPS and Section 4.3.1 of ANP-3676P for SPS describe the delayed reactor coolant pump (RCP) trip sensitivity studies. To ensure that the sensitivity studies were adequately performed to demonstrate compliance with 10 CFR 50.46, please provide the following for both NAPS and SPS:

a. The results of the break sizes that were analyzed for the hot leg and cold leg sensitivity studies in tables similar to those provided in Table 4-1 of ANP-3467P and ANP-3676P.

b. The results for the limiting break sizes for the cold leg and hot leg cases for NAPS and SPS in a table similar to those provided in Table 4-2 of ANP-3467P and ANP-3676P.

c. Discuss the modeling used for loop seal biasing in the hot leg and cold leg sensitivity studies and discuss if it was necessary to [ ] for the studies.

Response:

a. The results of the break sizes that were analyzed for hot and cold leg sensitivity studies are shown in Table 5.a-1 through 5.a-4 below:

Table 5.a-1

Summary of North Anna RCP Trip Delay Cold Leg SBLOCA Results

Table 5.a-1 (Continued)

Summary of North Anna RCP Trip Delay Cold Leg SBLOCA Results

Table 5.a-2

Summary of North Anna RCP Trip Delay Hot Leg SBLOCA Results

Table 5.a-2 (Continued)

Summary of North Anna RCP Trip Delay Hot Leg SBLOCA Results

Table 5.a-3

Summary of Surry RCP Trip Delay Cold Leg SBLOCA Results

Table 5.a-4

Summary of Surry RCP Trip Delay Hot Leg SBLOCA Results

b. The results for the limiting break sizes for the cold and hot leg cases are presented in Table 5.b-1 through 5.b-4 below:

Table 5.b-1

Sequence of events for North Anna RCP Trip Delay Cold Leg SBLOCA Study, 2.7-Inch, Limiting Case

Event	Time (sec)
Low PZR Pressure Trip	15
SIAS issued	25
RT and TT	17
HHSI Flow Loop 1	54
HHSI Flow Loop 2	54
HHSI Flow Loop 3	54
LHSI Flow Loop 1	-
LHSI Flow Loop 2	-
LHSI Flow Loop 3	-
SG 1 AFW	84
SG 2 AFW	-
SG 3 AFW	84
PCT Time	1932
LS 1 Clear	-
LS 2 Clear	-
LS 3 Clear	663
Break Uncovery	668
Core Uncovery	460
Accumulator Flow Loop 1	1906
Accumulator Flow Loop 2	1908
Accumulator Flow Loop 3	1906
Loss of Subcooling Margin	96
RCP Trip Time	396

Table 5.b-2

Sequence of events for North Anna RCP Trip Delay Hot Leg SBLOCA Study, 5.0-Inch, Limiting Case

Event	Time (sec)
Low PZR Pressure Trip	0.5
SIAS issued	11
RT and TT	2.5
HHSI Flow Loop 1	42
HHSI Flow Loop 2	42
HHSI Flow Loop 3	42
LHSI Flow Loop 1	-
LHSI Flow Loop 2	-
LHSI Flow Loop 3	-
SG 1 AFW	70
SG 2 AFW	-
SG 3 AFW	70
PCT Time	454
LS 1 Clear	280
LS 2 Clear	281
LS 3 Clear	281
Break Uncovery	270
Core Uncovery	292
Accumulator Flow Loop 1	440
Accumulator Flow Loop 2	440
Accumulator Flow Loop 3	440
Loss of Subcooling Margin	34
RCP Trip Time	334

Table 5.b-3

Sequence of events for Surry RCP Trip Delay Cold Leg SBLOCA Study, 3.0-Inch, Limiting Case

Event	Time (sec)
Low PZR Pressure Trip	0.8
SIAS issued	18
RT and TT (sec)	2.8
HHSI Flow Loop 1	58
HHSI Flow Loop 2	58
HHSI Flow Loop 3	58
LHSI Flow Loop 1	-
LHSI Flow Loop 2	-
LHSI Flow Loop 3	-
SG 1 AFW	72
SG 2 AFW	72
SG 3 AFW	72
PCT Time	1422
LS 1 Clear	2371
LS 2 Clear	2371
LS 3 Clear	539
Break Uncovery	546
Core Uncovery	414
Accumulator Flow Loop 1	1344
Accumulator Flow Loop 2	1344
Accumulator Flow Loop 3	1344
Loss of Subcooling Margin	80
RCP Trip Time	380

Table 5.b-4

Sequence of events for Surry RCP Trip Delay Hot Leg SBLOCA Study, 4.0-Inch, Limiting Case

Event	Time (sec)
Low PZR Pressure Trip	0.7
SIAS issued	14
RT and TT (sec)	2.7
HHSI Flow Loop 1	54
HHSI Flow Loop 2	54
HHSI Flow Loop 3	54
LHSI Flow Loop 1	-
LHSI Flow Loop 2	-
LHSI Flow Loop 3	-
SG 1 AFW	72
SG 2 AFW	72
SG 3 AFW	72
PCT Time	713
LS 1 Clear	274
LS 2 Clear	275
LS 3 Clear	276
Break Uncovery	355
Core Uncovery	357
Accumulator Flow Loop 1	1344
Accumulator Flow Loop 2	1344
Accumulator Flow Loop 3	1344
Loss of Subcooling Margin	49
RCP Trip Time	349

c. [

]

RAI 6

Request:

To assure the conservatism of the analysis used to demonstrate compliance with the limits of 10 CFR 50.46(b), please provide justification that the 5-minute trip delay time assumed in ANP-3467P (for NAPS) and ANP-3676P (for SPS) considers the limiting condition with respect to RCP operation for the full range of postulated SBLOCA scenarios. Please discuss considerations for larger breaks in the SBLOCA spectrum, as a 5-minute delay time is essentially equivalent to running the RCPs throughout the event, which has long been known to result in reduced peak cladding temperatures.

Please also address the likelihood that plant operators will take actions to trip the RCPs prior to the assumed 5-minute delay period. Alternatively, please perform additional sensitivity studies that consider reduced RCP trip time delays for break sizes 5 inches and larger for both NAPS and SPS to provide confidence that the limiting condition has been identified.

Response:

Framatome performed analysis cases for SPS and NAPS consistent with the treatment of RCP trip that is described in EMF-2328, Supplement 1 and its SER. The Framatome approach, as specified in Section 5.0 of EMF-2328, Supplement 1, is to reflect the existing plant basis for RCP trip, except where the evaluation model therein is used for relevant calculations.

According to the EMF-2328 Safety Evaluation,

"To prevent SBLOCAs from exceeding the criteria limits, the timing for tripping the RCPs during the event must also be identified. AREVA has agreed to evaluate a spectrum of hot and cold leg breaks to support the RCP trip procedure and determine/verify the trip timing consistent with the Emergency Operating Procedures (EOPs). This spectrum may include a sensitivity on RCP trip time if such is required to support the trip procedure or address an RAI from the NRC staff. The NRC staff accepts the AREVA proposed evaluation procedure for supporting the plant EOP for RCP trip timing following a LOCA."

With respect to the operator action time associated with manual RCP trip, the SBLOCA event at Three Mile Island Unit 2 on March 28, 1979 is referred to. As a result of this event, operation of the reactor coolant pumps (RCPs) during such transients was called into question. In the post-accident assessment, it was noted or hypothesized that several aspects of the event timeline affected the final core cooling outcome. Among these aspects was RCP trip. The morning of the event, the RCPs were left in operation for over an hour until cavitation due to highly voided suction conditions was causing severe pump vibration. This raised concerns that reactor coolant system (RCS) integrity could become challenged in that specific area and the pumps were ultimately shut-down. When the last RCPs were tripped approximately 1 hour and 40 minutes into the event, the fluid conditions in the RCS progressed rapidly from a quasi-homogenous saturated state, to a stratified one. This resulted in significant core uncovery due to a mass shortfall in the RCS.

The forensics of the event raised questions over this and many other design and operational aspects of pressurized water reactors (PWRs). These concerns prompted the NRC staff to issue many requests for action from the nuclear industry. This included directives for significant operator training improvements, control room staffing requirements, auxiliary feedwater system pedigree, new component design requirements and operation of the RCPs during RCS transients. Specific to operation of the RCPs, among the documents issued were NRC Bulletin 79-06C and Generic Letters 83-10 C and D. These documents presented questions for the need of automated RCP trip and/or improved guidance and operator training to facilitate proper RCP operation under various accident scenarios. Prior to this, Westinghouse had undertaken work to quantify the effects of RCP operation during SBLOCA transients. This work is documented in WCAP-9600 and more notably, WCAP-9584 References {6.1} and {6.2}.

In response to the NRC communications, the Westinghouse Owner's Group (WOG) issued several documents (OG-110 and OG-117, References {6.3} and {6.4}) which recommended RCP trip criteria on a generic basis. This work was partially based on analyses presented in WCAP-9584 and supplemented in OG-110 and OG-117. The analysis work itself was based on the WFLASH Evaluation Model (EM). The work in these documents strived to demonstrate the following:

a. The effects of longer term operation of the RCPs during SBLOCAs.

b. What the impacts of such operation may be on the SBLOCA licensing basis analysis.

c. Establish generic RCP trip criteria that could be provided in the emergency response guidelines, thus not necessitating an automated RCP trip under SBLOCA conditions.

Relative to the question being asked, the following conclusions were reached:

a. Automated RCP trip under SBLOCA conditions is not required.

b. Three RCP trip criteria were presented that could be utilized by the plant staff in the Emergency Operating Procedures (EOPs). They are as follows:

    1. An absolute RCS pressure with normal uncertainties
    2. Loss of hot leg sub-cooling
    3. Primary-to-secondary pressure differential

c. On a best estimate basis, the RCPs can be tripped at any time with acceptable SBLOCA results.

d. If the RCPs can remain operational throughout the entire small break transient, significant benefits relative to a maximum clad temperature occur due to enhanced steam cooling.

e. For any given break size, tripping the RCPs after the time when break flow becomes all steam (i.e., breakdown of two phase natural circulation/reflux cooling and progression to loop seal clearing) can make the SBLOCA results worse. The reason for this is that the break flow before that time remains a low-quality two-phase mixture which increases RCS mass loss with respect to time for a given pressure. The two main effects of RCP trip after this time are: 1) deeper core uncovery and 2) reduced total time of uncovery

(due to quicker accumulator injection). These two characteristics have opposing effects on PCT giving rise to a maximum function and a worst time interval of RCP trip.

f. As break size increases, the beneficial effects of reduced uncovery time from delayed RCP trip dominates any penalty from deeper core uncovery. The trend observed shows that as break size increases, the PCT penalty resulting from delaying the RCP trip decreases or vanishes.

g. Per WCAP-9584, when considering the spectrum of possible small break sizes, there exists a critical time such that, if RCPs are tripped no later than that time, PCTs will remain below 2200°F for that plant type regardless of the assumed break size. A 10 minute critical RCP trip time was determined for all Westinghouse NSSS designs. This was determined through an extensive analysis performed for the Westinghouse 3-Loop Plant which included many conservative analysis assumptions. In addition, the concept of an equivalent break size was utilized to conclude the critical time for 2-Loop and 4-Loop Plants. Therefore, this critical time can be applied on a generic basis. Note that in follow-up studies performed in Reference {6.4}, it was determined that the critical time could in some cases be as low as 5 minutes for a specific break size.

h. If the RCPs are tripped in conformance with the Westinghouse EOP Guidelines, the thermal-hydraulic system behavior and calculated peak clad temperature will be almost identical to the FSAR calculation assuming RCP trip at reactor trip time.

The main issue with RCPs running is that if the pumps remain in operation too long, additional mass loss from the reactor coolant system can be expected as compared to cases where the RCPs are tripped. This is because the break flow quality can remain relatively low for an additional operational period which will increase mass loss for a given RCS pressure. However, if the RCPs can be tripped before system mass loss is equivalent to the liquid phase inventory that remains after the loop piping has drained (post-loop seal clearing), the SBLOCA transient results are very similar between RCPs operating vs. RCPs tripped. The transient event sequence timing may shift, however, the overall RCS response remains very similar. This is not necessarily based on code results, but rather physics confirmed by the system codes.

For a cold leg SBLOCA with loss of off-site power, the RCS response is generally as follows. When RCPs trip due to loss off-site power from turbine trip, the SBLOCA transient will progress to a single phase natural circulation period which maintains core cooling. As saturation is reached due to mass loss and subsequent depressurization, natural circulation will transition to a two-phase state. When over-all mass loss exceeds approximately 40% (note this will vary somewhat depending on break size and decay heat power), the relative velocity between the liquid and vapor phases becomes too great in the steam generator (SG) vertical tube runs to support co-current flow. At that point the two-phase mixture circulation breaks down into a counter-current reflux cooling period. As mass loss progresses, the liquid trapped in the RCP suction cross-over legs is purged and venting of the vapor being generated by the core begins. At this time, the amount of liquid inventory that remains in the system following initial loop seal clearing is basically confined to the vessel.

As break size increases, the time to loop seal clearing is reduced, though the time rate of change on RCS pressure becomes larger which reduces break flow, increases emergency core cooling system (ECCS) flow and leads to earlier accumulator injection all of which minimize the duration of core uncovery. Therefore the effect of continued RCP operation becomes diminished for larger breaks.

The time rate of change of RCS pressure for a given break size can be impacted by RCPs running depending on when the pumps are tripped and how much mass remains in the system at time of trip. This variation is not extreme though. Again, as long as the RCPs are tripped at a point before loop seal clearing would occur in the loss of off-site power cases, the operation of the pumps up to that time will have little effect on the analysis. These are all physical phenomena which are considered independent of thermal-hydraulic system codes and models. Therefore, regardless of the change in code used, the basis of the WOG generic RCP trip criteria is upheld. That is, for the cases of significance in the SBLOCA analysis, the RCPs will be tripped in a time frame before the transient is adversely impacted. If the trip is not performed in a timely manner with regard to the transient, that is, the criteria exists almost instantaneously to tripping the RCPs, the break size is large enough such that the impact of operating RCPs will be minimal because of the significant depressurization that would have occurred and the accompanying benefits of such phenomena (i.e. earlier actuation of safety systems resulting in reduced time of core uncovery). Note that this rationale applies to hot leg breaks as well. That is, until the liquid phase inventory reaches the break elevation, the differences in cases between RCPs in operation and those where off-site power is lost at turbine trip, will essentially be the same. This is due to the break donor quality remaining basically the same during that time frame, i.e., at or near a saturated liquid state. There could be some subtle differences in break flow due to differences in depressurization rates, but these are not considered to cause major differences in system mass loss with respect to time.

Surry and North Anna Emergency Operating Procedures for response to Loss of Coolant Accidents contain continuous action steps that direct operators to stop all RCPs upon indication of loss of RCS subcooling. Dominion maintains an engineering basis document which lists this action as a key operator action, with a completion time of 5 minutes. Operator training includes recurrent activities to validate the accomplishment of these actions within the stated completion time.

For application of the FVI-SBLOCA methodology, the existing Surry and North Anna basis for RCP trip has been validated. The existing basis relies upon the generic Westinghouse Owner's Group analysis and RCP trip setpoint recommendations which were approved by the NRC in Generic Letter 85-12. This combination of design analyses and EOP action provides assurance that the limits of 10 CFR 50.46(b) will be met, consistent with the intent stated in the NRC's Safety Evaluation contained in Generic Letter 85-12.

References:

{6.1} WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System," June 1979.

{6.2} WCAP-9584, "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss of Coolant Accidents for Westinghouse Nuclear Steam Supply Systems," August 1979.

{6.3} Westinghouse Owner's Group Letter, OG-110, "Evaluation of Alternate RCP Trip Criteria," October 6, 1983.

{6.4} Westinghouse Owner's Group Letter, OG-117, "Justification of Manual RCP Trip for Small Break LOCA Events," March 9, 1984.

RAI 7

Request:

The NRC staff's SE for EMF-2328(P)(A), Revision 0, Supplement 1, [ ] An additional source of safety injection is the containment sump. Should the refueling water storage tank (RWST) be drained down before the core is quenched, the switchover to the containment sump will result [ ], which could potentially cause a more bounding PCT for the limiting break size later in the analysis.

To ensure the PCT for the limiting break is conservatively determined to demonstrate compliance with 10 CFR 50.46(b) acceptance criteria, please indicate whether an RWST drain down analysis has been performed for the limiting break size and identify the value used [ ] Please justify how this value is acceptably bounding.

Response:

An RWST drain down analysis has been performed to account for the effect of the drain down of the RWST [ ] The results and conclusions of this sensitivity study support that the analyses of record results as presented in the submittals remain bounding.

Once the minimum usable water volume of the RWST has been exhausted due to the combination of injected SI and containment spray flow, [ ]

RAI 8

Request:

As discussed in the license amendment request (LAR), the FVI SBLOCA analysis methods deviate from the approved EMF-2328(P)(A) methodology. [

] To ensure that the methodology continues to appropriately predict a SBLOCA transient to demonstrate compliance with 10 CFR 50.46(b) acceptance criteria, please provide the following:

a. [ ]

b. [ ]

c. [ ]

Response:

[

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[

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RAI 9

Request:

Regarding the SPS FVI SBLOCA Analysis, Table 3-1 of ANP-3637P (later clarified to be ANP-3676P) contains a value of 1899.7 psia (pounds per square inch absolute) for the pressurizer pressure - low reactor trip setpoint. However, the SPS technical specifications contain a value of ≥1875 psig (pounds per square inch gauge) (TS 2.3.A.2(c)). A lower value is traditionally more conservative for a SBLOCA since it will increase the time for the reactor to trip after the break. Please discuss if the TS value was included in the analysis and, if not, justify that the use of the higher setpoint is conservative.

Response:

The value listed in Table 3-1 of ANP-3676P is the nominal setpoint for the low pressurizer pressure reactor trip. The value used in the SPS FVI-SBLOCA analysis for the low pressurizer pressure reactor trip setpoint is **[**

] Therefore, even though the Nominal low pressurizer pressure reactor trip setpoint of 1899.7 psia was listed in ANP-3676P, **[**

]

RAI 10

Request:

The plant-specific methodology reports ANP-3467P (for NAPS) and ANP-3676P (for SPS) suggest that a reactor trip on low pressurizer pressure is applicable to the full range of analyzed small breaks (i.e., 1 to 8.7 inches). [

]

a. [

]

b. [

]

c. [

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Response:

a. The low pressurizer pressure RPS trip is based on a compensated pressurizer pressure signal. Pressurizer pressure is compensated by a lead-lag controller. The trip function is diagrammed in the North Anna UFSAR (see Figure 7.2-6). Surry is configured likewise. This is an anticipatory signal and would result in a trip quite a bit earlier than if it were based on a straight pressurizer pressure measurement.

[

]

b. [

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[

]

c. [

]

RAI 11

Request:

Section 3.2.2 of ANP-3676P states that, [

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Response:

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- [

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[

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- [

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- [

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- [

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[1] M5Framatome is a registered trademark of Framatome or its affiliates, in the USA or other countries.

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Table 11-1

[

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Table 11-2

RAI 12

Request:

The FVI SBLOCA analysis methodology postulates that generic fuel modeling characteristics may be used in a SBLOCA analysis provided that a simplified set of fuel performance criteria is satisfied. For NAPS, the licensee stated the analysis is applicable to 17x17 fuel products with ZIRLO and Optimized ZIRLO cladding per ANP-3467P (a similar statement is made for SPS in ANP-3676P, which uses a 15x15 lattice in lieu of a 17x17 lattice). However, the NRC staff observed documents that discussed [

] could be relevant since they may affect fuel steady-state and transient behavior. However, these parameters were not explicitly addressed in the application. Therefore, to assure the actual plant design is applicable to the analysis used to demonstrate compliance with the limits of 10 CFR 50.46(b), please provide a detailed description of and justification for the fuel-related parameters that must be satisfied for:

a. The specific analyses described in ANP-3467P (for NAPS) and ANP-3676P (for SPS) to be considered applicable for a given fuel cycle.
b. The analytical methods described in EMF-2328(P)(A), ANP-3467P (for NAPS), and ANP-3676P (for SPS) to be considered applicable for a new fuel design.

Response:

a. [

]

[

]

b. [

]

RAI 13

Request:

The FVI methodology proposed by the licensee does not appear to include explicit requirements for data quality and quantity to support assessment of the EM to new fuel types that are different from those that have been explicitly evaluated (e.g., fuel designs clad with ZIRLO and Optimized ZIRLO). For example, it is possible that the licensee could seek to apply the FVI model to fuel that is presumed to satisfy specific criteria, despite a lack of available data (e.g., for swelling and rupture) to validate EM applicability. In other cases, only a few data points may be available, which may be insufficient to characterize the domain of concern. To assure the applicability and conservatism of the EM used to demonstrate compliance with the limits of 10 CFR 50.46(b), please clarify the applicability of the FVI methodology in cases where limited or no data is available to validate that the EM is applicable to a certain fuel design.

Response:

The response is based on a presumption that available data for cladding and fuel rod characteristics are limited to support application of the EM to new fuel types that are different from those that have been explicitly evaluated. Applicability of the FVI methodology is not established presuming that criteria are met, as stated in the request. FVI application requires exercising the established process to determine whether the characteristics of an alternate cladding material may be represented by the models included in the FVI analysis. This process will be employed for any potential alternate fuel design and cladding material. The process identifies certain characteristics as necessary to confirm FVI applicability to an alternate material: **[**

] Available data from the cognizant fuel vendor or other sources would be employed to assess whether the data are representative of data used in the FVI analysis models. For a cladding product that a fuel vendor considers potentially viable for general use, it is expected that sufficient data are likely to be available to perform the assessment.

It is possible that limited data may exist for a cladding product under consideration for application of the FVI methodology. In this unlikely situation, other actions may be taken, such as performing sensitivity studies, or defining a conservative PCT penalty to compensate for limited data. These potential actions, including the application and magnitude of such a penalty, would be determined based on the specific product under consideration. This approach provides additional conservatism for the application of the FVI-SBLOCA methodology to the alternate fuel design. This overall approach represents a conservative treatment to demonstrate compliance with the provisions of 10 CFR 50.46.

RAI 14

Request:

Considering, in particular, the intent of requirements expressed in 10 CFR 50.59 and associated regulatory guidance, please provide clarification and justification as to when prior review by the NRC staff would be necessary to support application of the methods described in EMF-2328(P)(A) and ANP-3467P (for NAPS) or ANP-3676P (for SPS) to a new fuel design that has not been previously analyzed with these methods.

Response:

Dominion procedures incorporate the provision in 10 CFR 50.59(c)(4), which specifically excludes from the scope of 10 CFR 50.59 changes to the facility or procedures that are controlled by other more specific requirements and criteria established by regulation. Application of the FVI-SBLOCA methodology to a different fuel design would be treated as a change in the application of an evaluation model, as specified in 10 CFR 50.46(a)(3)(ii). For such an application, the governing process is 10 CFR 50.46 instead of 10 CFR 50.59.

Dominion has developed a process to evaluate other fuel design and cladding material properties with respect to the applicability of the FVI-SBLOCA analyses. When validated by this process, the impact of fuel product changes on the SBLOCA analysis is assessed and evaluated for reportability under the provisions of 10 CFR 50.46, as noted above. Dominion procedures for 10 CFR 50.46 reporting are consistent with NRC Regulatory Issue Summary 2016-04. Specifically, a fuel product change would be evaluated as a change to the ECCS EM and evaluated for reportability pursuant to 10 CFR 50.46.

In situations where the change to the facility involves aspects that are not fully governed by 10 CFR 50.46, those aspects would be considered under the provisions of 10 CFR 50.59. If the change to the facility was found to require NRC review under provisions of 10 CFR 50.59, then the necessary evaluations would be provided to support the NRC review of the requested change.

RAI 15

Request:

The NRC staff understands that the licensing basis steady-state fuel performance modeling will not necessarily be performed by the EM furnisher, but that fuel performance inputs will be generated, in accordance with Framatome methods, using RODEX2-2A. Additional information is required to determine whether the fuel rod modeling process will remain conformant to the requirements associated with initial stored energy, swelling and rupture of the cladding, and fuel rod thermal parameters, specified in Appendix K to 10 CFR 50.

a. Provide a summary of the process used to supply information to Framatome, in order to generate the necessary initial conditions in RODEX2-2A.

b. Describe in detail how the fuel rod is initialized, i.e., initial conditions generated using RODEX2-2A and then supplied to S-RELAP5, for the SPS and NAPS plants, and summarize the specific data transferred between RODEX2-2A and S-RELAP5.

c. Explain how the licensee ensures that the EM is initialized using fuel performance data that are reflective of and appropriate for the cycle design characteristics that exist at either plant.

Response:

a. Dominion provided Framatome with fuel and plant inputs to be used in the FVI-SBLOCA analyses for North Anna and Surry, including the inputs to generate the necessary initial conditions in RODEX2-2A. The code is used to generate initial conditions (fuel-to-clad gap dimensions, pin internal gas composition and pressure) of the fuel pin as a function of burnup. [

]

The inputs Dominion provided Framatome for RODEX2-2A initialization include:

- General fuel design parameters:

  General fuel parameters of the resident fuel products that are not Westinghouse-proprietary were provided to Framatome for use in the FVI-SBLOCA analyses. This includes items such as fuel lattice, number of assemblies, control rod guide tube inside/outside diameters etc. [

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b. Framatome SBLOCA EM fuel rod initial and transient conditions are generated using the standalone RODEX2-2A code and the RODEX2-2A fuel model built into S-RELAP5 (S-RELAP5/RODEX2-2A model), respectively. The fuel rod physical phenomena treatment is divided into two categories:

   1. [

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   2. [

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c. The parameters that require cycle-to-cycle disposition to ensure that the FVI-SBLOCA analyses for North Anna and Surry are reflective of and appropriate for the cycle design characteristics that exist at either plant are discussed in the response to Part a of RAI 12. With respect to the inputs Dominion provided to Framatome for RODEX2-2A initialization as discussed in Part a of this RAI, the parameter that is to be checked on a reload cycle basis is the Axial Power Shape. The other inputs are not expected to be changed with cycle design variations and, hence, are not included in the reload check.

RAI 16

Request:

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Response:

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The SBLOCA break spectrum includes break sizes up to 10% of the cold leg area. [

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References:

{16.1} Safety Evaluation, U.S. EPR DC PSER, P2 Group II, (including 15.6.5.1 thru 3 but not including 15.6.5.4), Docket Number 05200020, (NRC Adams Accession Number ML14358A163).

{16.2} EMF-2328(P)(A), Revision 0, Supplement 1 (P)(A), Revision 0, PWR Small Break LOCA Evaluation Model, S-RELAP5 Based.

RAI 17

Request:

Discussion in ANP-3467P and ANP-3676P states that the FVI analysis explicitly models a Framatome fuel design that is "functionally very similar" to other vendor fuel designs that may be loaded at NAPS and SPS. This statement suggests that the FVI analysis applicability would require functional similarity for fuel designs loaded at these reactors. Presumably, this would include fuel cladding that is a zirconium-based alloy. To assure applicability of the analysis used to demonstrate compliance with the limits of 10 CFR 50.46(b) to the actual plant design, specify and provide justification for all functional requirements associated with the cladding material (e.g., minimum zirconium content, concentrations of other alloying agents) and other characteristics deemed necessary for a fuel design to be considered "functionally very similar" as described in ANP-3467P and ANP-3676P.

Response:

The request highlights the phrase "functionally very similar" in a manner that implies it represents a requirement or acceptance criterion for application of the FVI-SBLOCA to an alternate fuel product. The phrase only represents a general characterization of features of the Framatome fuel assembly modeled in the FVI-SBLOCA analysis, as compared with the resident fuel. Other discussion contained in Section 3.2.2 of ANP-3467P and ANP-3637P describes [

] The phrase "functionally very similar" is not intended to convey any additional or different characteristics than those specifically addressed in ANP-3467P and ANP-3637P.

Dominion has developed a process to evaluate the applicability of the FVI-SBLOCA analyses to a new fuel product that has the same fuel lattice as the current resident fuel and uses an approved Zirconium-based alloy cladding material. This process, discussed in Part b of RAI 12, represents the engineering guidance for assessment of the specific relevant fuel design characteristics.

RAI 18

Request:

The plant-specific methodology reports ANP-3467P (for NAPS) and ANP-3676P (for SPS) identify that the existing SBLOCA EM described in EMF-2328(P)(A) is applicable to fuel designs using ZIRLO and Optimized ZIRLO cladding. [

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Response:

The applicable regulatory requirements related to clad deformation (ballooning) and rupture applicable to the SBLOCA analysis are stated in Appendix K to 10 CFR 50, Section I.B, which requires that the swelling and rupture calculations to be "based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated". Due to its influence on the PCT and local oxidation this phenomenon will be given special attention.

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Table 18-1

Table 18-2

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Figure 18-1

Figure 18-2

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References:

{18.1} [
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{18.2} [
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{18.3} [
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{18.4} [
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{18.5} EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based", March 2001.

{18.6} EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based", December 2016.

{18.7} [
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RAI 19

Request:

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Response:

Appendix K requires that the degree of swelling and incidence of rupture are not underestimated. The occurrence of rupture has a high importance during the boiloff period of an SBLOCA event. Once rupture occurs, the local balloon size is significantly increased relative to that from the stress-induced swelling and unoxidized cladding is exposed to steam in the coolant channel. The Baker-Just correlation, prescribed by Appendix K, is used to calculate the subsequent interior oxidation growth and the corresponding heat added to the cladding.

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Table 19-1

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Table 19-2

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Table 19-3

Table 19-4

Table 19-5

Table 19-6

Figure 19-1

RAI 20

Request:

To assure the robustness and conservatism of the EM used to demonstrate compliance with the limits of 10 CFR 50.46(b), consider a sensitivity study using RODEX2-2A and S-RELAP5 that explicitly analyzes a recent Framatome fuel design (e.g., GAIA) and an older design with significant differences (e.g., Advanced Mark-BW) to validate that the theoretical arguments presented in the license amendment request are valid for a real-world demonstration case. Please demonstrate that the differences observed in the sensitivity analysis (e.g., consider 3-in and 6.5-in break sizes) are insignificant or otherwise justify that they do not impact the conservatism of the EM.

Response:

As discussed in the responses to RAIs 11 and 12, validation of the EM with FVI application relies on the [

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and sufficiently demonstrate compliance with the limits of 10 CFR 50.46(b).

RAI 21

Request:

To demonstrate that the FVI SBLOCA analysis would be capable of conservatively representing fuel designs clad with other zirconium-based alloys when demonstrating compliance with the limits in 10 CFR 50.46(b), please provide an S-RELAP5 sensitivity study for the 3-in and 6.5-in break size cases comparing the M5-proprietary cladding models to those applicable to Zircaloy (e.g., NUREG-0630 swelling and rupture models).

Please justify that the differences observed are insignificant or otherwise justify that they do not impact the conservatism of the EM.

Response:

A sensitivity study comparing Zirc-4 cladding models to the $M5_{Framatome}$ cladding models used in the licensed Millstone Unit 2 SBLOCA analysis is described in Reference {21.1}. Changes made to the inputs between $M5_{Framatome}$ and Zirc-4 include [

]

The break sizes and associated PCTs for the Zirc-4 cladding evaluation are listed in the following table.

Summary of Results for the Zr-4 SBLOCA Break Spectrum

Break Diameter (in)	3.76	3.78	3.785	3.79	3.90	4.00
PCT (°F)	1646	1711	1682	1541	1586	1677

Comparing these PCT values to the limiting PCT calculated for $M5_{Framatome}$ (1707°F) results in a calculated PCT delta of 4 degrees Fahrenheit between the limiting cases. [

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References

{21.1} Letter from Mark D. Sartain to U.S. Nuclear Regulatory Commission (Serial No. 16-070), "Dominion Nuclear Connecticut, Inc., Millstone Power Station Unit 2, Response To Request For Additional Information For Proposed License Amendment Request Regarding Small Break Loss Of Coolant Accident Reanalysis (CAC No. MF6700)," dated March 24, 2016 (Accession Number ML16096A388).

RAI 22

Request:

ANP-3467P and ANP-3676P did not provide validation to assure that conceivable differences in fuel design would not adversely influence the conservatism of the figures of merit intended to satisfy acceptance criteria in 10 CFR 50.46(b). These differences in fuel design do not appear to be explicitly accounted for in assessing the applicability of the SBLOCA analysis and analytical methods to new fuel cycles and fuel designs.

Perform a sensitivity study for the 3-in and 6.5-in break size cases using RODEX2-2A and S-RELAP5 to demonstrate that conceivable changes in non-evaluated parameters deemed to be of secondary importance [

] have a minor effect on the progression of the SBLOCA transient and its results.

For this study, hold the fuel rod design and cladding parameters the licensee deems to be of primary importance in its analysis and EM assessment procedures constant.

Response:

As discussed in the responses to RAI 11 and 12, validation of the EM with FVI application [

] Thus, the fuel design parameters that have significant impact on transient response and subsequently the acceptance criteria are shown to be modeled conservatively in the EM and sufficiently demonstrate compliance with the limits of 10 CFR 50.46(b).

RESOLUTION OF SBLOCA ANALYSIS ERROR

The purpose of this section is to address an SBLOCA analysis error for North Anna and Surry Power Stations and provide the correct [ ] results to the NRC to support their review of LAR for these plants.

NORTH ANNA [ ] CASE

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Table 1

Summary of North Anna SBLOCA Results – [ ] Case

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Table 2

North Anna Event Times – [ ] Case

Figure 1

North Anna Break Spectrum – PCT Result

SURRY [ ] CASE:

[

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Table 3

Summary of Surry SBLOCA Results – [ ] Case

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Table 4

Surry Event Times – [ ] Case

Figure 2

Surry Break Spectrum – PCT Result

Attachment 3

REVISED MARKED-UP NAPS AND SPS TECHNICAL SPECIFICATIONS PAGES

**Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna and Surry Power Stations Units 1 and 2**

3. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," as supplemented by ANP-3467P, "North Anna Fuel-Vendor Independent Small Break LOCA Analysis," as approved by NRC Safety Evaluation Report dated [DATE].

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology."

2. Plant-specific adaptation of WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," as approved by NRC Safety Evaluation Report dated February 29, 2012.

~~3. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."~~

~~4. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."~~

4. ~~5.~~ WCAP-12610, "VANTAGE+ FUEL ASSEMBLY—REFERENCE CORE REPORT."

5. ~~6.~~ VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."

6. ~~7.~~ VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

7. ~~8.~~ WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."

8. ~~9.~~ WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."

9. ~~10.~~ BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel."

10. ~~11.~~ BAW-10199P-A, "The BWU Critical Heat Flux Correlations."

11. ~~12.~~ BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

12. ~~13.~~ EMF-2103 (P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."

13. ~~14.~~ EMF-96-029 (P)(A), "Reactor Analysis System for PWRs."

~~15. BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II only (SBLOCA models).~~

14. ~~16.~~ DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix A, "Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code," Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code."

15. ~~17.~~ WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO" (Westinghouse Proprietary).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

3. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," as supplemented by ANP-3676P, "Surry Fuel-Vendor Independent Small Break LOCA Analysis," as approved by NRC Safety Evaluation Report dated [DATE].

The analytical methods used to determine the core operating limits identified above shall be those previously reviewed and approved by the NRC, and identified below. The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number, title, revision, date, and any supplements). The core operating limits shall be determined so that applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided for information for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REFERENCES

1. VEP-FRD-42-A, “Reload Nuclear Design Methodology”
2. WCAP-16009-P-A, “Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),” (Westinghouse Proprietary).
~~3. WCAP-10054-P-A, “Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code,” (W Proprietary)~~
~~4. WCAP-10079-P-A, “NOTRUMP, A Nodal Transient Small Break and General Network Code,” (W Proprietary)~~

4. ~~5.~~ WCAP-12610-P-A, “VANTAGE+ Fuel Assembly Report,” (Westinghouse Proprietary)

5. ~~6.~~ VEP-NE-2-A, “Statistical DNBR Evaluation Methodology”

6. ~~7.~~ VEP-NE-3-A, “Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code”

7. ~~8.~~ DOM-NAF-2-A, “Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code,” including Appendix B, “Qualification of the Westinghouse WRB-l CHF Correlation in the Dominion VIPRE-D Computer Code,” and Appendix D, “Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code”

8. ~~9.~~ WCAP-8745-P-A, “Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function”

9. ~~10.~~ WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, “Optimized ZIRLO,” (Westinghouse Proprietary)

Attachment 4

REVISED TYPED NAPS AND SPS TECHNICAL SPECIFICATIONS PAGES

Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna and Surry Power Stations Units 1 and 2

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology."

2. Plant-specific adaptation of WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," as approved by NRC Safety Evaluation Report dated February 29, 2012.

3. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," as supplemented by ANP-3467P, "North Anna Fuel-Vendor Independent Small Break LOCA Analysis," as approved by NRC Safety Evaluation Report dated [DATE].

4. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY—REFERENCE CORE REPORT."

5. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."

6. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

7. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."

8. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."

9. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel."

10. BAW-10199P-A, "The BWU Critical Heat Flux Correlations."

11. BAW-10170P-A, "Statistical Core Design for Mixing Vane Cores."

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

12. EMF-2103 (P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."

13. EMF-96-029 (P)(A), "Reactor Analysis System for PWRs."

14. DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix A, "Qualification of the F-ANP BWU CHF Correlations in the Dominion VIPRE-D Computer Code," Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code."

15. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO" (Westinghouse Proprietary).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

The analytical methods used to determine the core operating limits identified above shall be those previously reviewed and approved by the NRC, and identified below. The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number, title, revision, date, and any supplements). The core operating limits shall be determined so that applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided for information for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REFERENCES

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology"
2. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary).
3. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model S-RELAP5 Based," as supplemented by ANP-3676P, "Surry Fuel-Vendor Independent Small Break LOCA Analysis," as approved by NRC Safety Evaluation Report dated [DATE].
4. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Report," (Westinghouse Proprietary)
5. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology"
6. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code"
7. DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix B, "Qualification of the Westinghouse WRB-1 CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code"
8. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function"
9. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO," (Westinghouse Proprietary)