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August 31, 2007

TVA-BFN-TS-418
TVA-BFN-TS-431

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop OWFN, P1-35
Washington, D. C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) – UNITS 1, 2, AND 3 - TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418 – EXTENDED POWER UPRATE (EPU) – RESPONSE TO ROUND 13 REQUEST FOR ADDITIONAL INFORMATION (RAI) - CONTAINMENT OVERPRESSURE – (TAC NOS. MD5262, MD5263, AND MD5264)

By letters dated June 28, 2004 (ADAMS Accession No. ML041840109) and June 25, 2004 (ML041840301), TVA submitted license amendment applications for EPU of BFN Unit 1 and BFN Units 2 and 3, respectively. On July 5, 2007, the NRC staff issued a Round 13 RAI (ML071780190) regarding the EPU license amendment requests. The Round 13 RAI contains a set of APLA RAIs (containment overpressure) and seven additional SBWB RAIs. TVA's responses to the Round 13 SBWB RAI items were submitted on August 9, 2007 (ML072270037). The enclosure to this letter provides TVA’s responses to all but one of the remaining Round 13 containment overpressure RAI items. As discussed with the NRC staff, TVA is preparing the response to the remaining RAI (APLA-35/37).

TVA has determined that the additional information provided by this letter does not affect the no significant hazards considerations associated with the proposed TS changes. The proposed TS changes still qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

No new regulatory commitments are made in this submittal. If you have any questions regarding this letter, please contact me at (256)729-3612.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 31st day of August, 2007.

Sincerely,

Original signed by:

D. T. Langley
Manager of Licensing
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Enclosure:

Response to Round 13 Request For Additional Information - Containment Overpressure

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ENCLOSURE

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418
EXTENDED POWER UPRATE (EPU)

RESPONSE TO ROUND 13 REQUEST FOR ADDITIONAL INFORMATION
CONTAINMENT OVERPRESSURE

NRC Request APLA-28/30

Confirm that the following success criteria were used to estimate the risk related to the containment overpressure (COP) credit:

Initiator	Injection	NPSH
Large Break Loss of coolant Accident (LLOCA)	1 core spray (CS) pump or 1 Low Pressure Core Injection (LPCI) pump	3 or 4 residual heat removal (RHR) pumps/heat exchanges (H/Xs) aligned to spent fuel pool cooling (SPC) or 2 RHR pumps/HXs aligned to SPC and favorable plant conditions (initial SP volume at 123,500 ft^3, river water temperature at 85 degrees F, torus water temperature less than or equal to 86 degrees F) or 2 pumps/HXs aligned to SPC and containment integrity (COP credit) Note: 1 pump/HX aligned to SPC and containment integrity (COP credit) will not provide adequate NPSH to the low pressure emergency core cooling system (ECCS) pumps during LLOCAs

Anticipated Transient Without Scram (ATWS) or Station Blackout (SBO) (upon AC power recovery after 4 hours)	1 CS pump or 1 LPCI pump	containment integrity (COP credit) Note: Does not depend on the number of RHR pumps/HXs aligned to SPC
Other transients	1 CS pump or 1 LPCI pump	2 or more RHR pumps/HXs aligned to SPC or 1 pump/HX aligned to SPC and containment integrity (COP credit)

TVA Reply to APLA-28/30

The risk estimates provided in response to RAI APLA-24/26 in the July 21, 2006 submittal (ML062090071) and APLA-26/28 in the September 15, 2006 submittal were based on the BFN probabilistic risk assessment (PRA) model which credits numerous plant systems for core cooling and decay heat removal in addition to those in the table which take suction from the suppression pool.

The PRA models used in the risk estimates were reviewed for success criteria and corrections/comments to the table provided in the RAI request are shown below. Additions are indicated with **bold double underline** and deletions are indicated as ~~strikeout~~.

Initiator	Injection	NPSH
Large Break Loss of coolant Accident (LLOCA)	1 core spray (CS) **loop** ~~pump~~ or 1 Low Pressure Core Injection (LPCI) pump	3 or 4 residual heat removal (RHR) pumps/heat exchanges (H/Xs) aligned to **suppression** ~~spent fuel~~ pool cooling (SPC) **or** **2 RHR pumps/HXs aligned to SPC and favorable plant conditions (102% power, 2 sigma decay heat,initial SP volume at 123,500 ft^3, river water temperature at 68 degrees F, torus water temperature less than or equal to 87 degrees F)** or

		2 RHR pumps/HXs aligned to SPC <u>and</u> favorable plant conditions (**<u>100% power, nominal decay heat,</u>** initial SP volume at 123,500 ft^3, river water temperature at 85 degrees F, torus water temperature less than or equal to 86 degrees F) <u>or</u> 2 pumps/HXs aligned to SPC <u>and</u> containment integrity (COP credit) Note: 1 pump/HX aligned to SPC <u>and</u> containment integrity (COP credit) **<u>was not considered or credited for</u>** ~~will not provide adequate~~ NPSH to the low pressure emergency core cooling system (ECCS) pumps during LLOCAs
Anticipated Transient Without Scram (ATWS) <u>or</u> Station Blackout (SBO) (upon AC power recovery after 4 hours)	1 CS pump <u>or</u> 1 LPCI pump	containment integrity (COP credit) Note: Does not depend on the number of RHR pumps/HXs aligned to SPC
Other transients	1 CS pump <u>or</u> 1 LPCI pump	2 or more RHR pumps/HXs aligned to SPC ~~<u>or</u>~~ ~~1 pump/HX aligned to SPC <u>and</u> containment integrity (COP credit)~~

<u>NRC Request APLA-29/31</u>

Describe any interlocks or procedural prohibitions that preclude the simultaneous opening of the LPCI valves, SPC valves, and/or drywell spray valves in the same RHR subsystem.

TVA Reply to APLA-29/31

The suppression pool cooling/spray valves and the drywell spray valves are interlocked closed via control circuit logic any time the LPCI initiation logic is satisfied. This design feature ensures that LPCI flow is directed to the reactor and is not diverted to the suppression pool cooling/spray or drywell spray functions. These interlocks may be manually overridden or bypassed through the use of handswitch controls in the main control room. The use of these handswitches is governed by plant procedures.

Assuming manual alignment by the operators, and recognizing the use of the bypass handswitches, it is physically possible to have LPCI, suppression pool cooling/spray, and/or drywell spray valves in the open position within the same RHR subsystem at the same time. Procedures prohibit the use of any containment cooling flowpath in conjunction with the LPCI flowpath within a single RHR loop. Multiple containment cooling flowpaths (i.e., suppression pool cooling, suppression pool spray, and/or drywell spray) within a single RHR loop are allowed by BFN procedures.

NRC Request APLA-30/32

The licensee has made a commitment to terminate drywell cooling within two hours of entry into the Appendix R fire safe shutdown operating procedures. Address how this commitment was considered during development of the probabilistic risk assessment (PRA) success criteria.

TVA Reply to APLA-30/32

Operation of the drywell coolers during the Design Basis Accident - LOCA (DBA-LOCA), Appendix R, ATWS, and SBO events was discussed previously in the response to RAI ACVB.41/39 in the submittal dated August 4, 2006 (ML062220647).

TVA provided an evaluation of additional PRA sequences in response to RAI APLA-26/28 in Enclosure 2 of the September 15, 2006 submittal. This evaluation included events (other than LOCA and ATWS) leading to either depressurization using safety relief valves (SRV) upon loss of high pressure make-up or multiple, stuck open SRVs to quantify the impact on PRA results if ECCS pump net positive suction head (NPSH) and COP are considered. An upper bound frequency of those scenarios resulting from general transient scenarios was determined. The results of the evaluation determined that based on the upper bound delta frequency, consideration of COP for these events is non-risk significant. Based on the low frequency result of the evaluation, it was not necessary to further consider the risk associated with COP credit (or drywell cooling operation) for these events.

NRC Request APLA-31/33

The SPC mode is manually aligned by the operator. Describe how the operator decides the number of RHR pumps/HXs to align to SPC. Address whether it is credible (e.g., within procedural guidance) that the operator would actually align three or four RHR pumps/HXs to SPC.

TVA Reply to APLA-31/33

Suppression pool temperature greater than 95°F is an entry condition for Emergency Operating Instruction (EOI)-2, "Primary Containment Control." EOI-2 directs the operator to monitor and control the suppression pool temperature below 95°F using available suppression pool cooling; and, if a pool temperature less than 95°F cannot be maintained, Step SP/T-3 of the EOI specifically directs the operator to operate all available RHR pumps for suppression pool cooling that are not required to assure adequate core cooling by continuous injection.

NRC Request APLA-32/34

Determine if there is a significant statistical correlation between the suppression pool water level, river water temperature, and/or torus water temperature.

TVA Reply to APLA-32/34

The "BFN EPU Containment Overpressure (COP) Credit Risk Assessment," (provided as Enclosure 2 to the July 21, 2006 submittal) included evaluations of suppression pool water level, suppression pool temperature, and river water temperature historical data. This data has been reviewed and no discernable correlation could be identified between suppression pool level and either suppression pool temperature or river water temperature.

NRC Request APLA-33/35

Discuss whether it is possible to eliminate or substantially reduce the need for the COP credit by maximizing the suppression pool water level, minimizing the initial torus water temperature and/or reducing the amount of power uprate.

TVA Reply to APLA-33/35

The BFN units require COP credit under certain conditions for both the originally licensed power level of 3293 megawatt thermal (MWt) and for the current licensed power level of 3458 MWt. The need for COP credit is not being introduced to the BFN design basis via the EPU request currently under evaluation.

Technical Specification 3.6.2.2, "Suppression Pool Water Level," specifies the acceptable band of suppression pool level during power operations as ≥ -6.25 inches with and -7.25 inches without differential pressure control and ≤ -1.0 inches. If the suppression pool water level is too high, excessive clearing loads from SRV discharges and excessive pool swell loads during a DBA LOCA could result. Ambient, seasonal weather conditions limit the ability to minimize the initial suppression pool water temperature because the suppression pool is cooled using the river as the heat sink. There are no practical ways to increase suppression pool level or decrease the suppression pool temperature sufficiently to have significant impact on the need for COP credit.

NRC Request APLA-34/36

Provide the approximate high confidence of low probability of failure (HCLPF) of the Browns Ferry containment structure (including considerations of personnel access or equipment hatches, penetrations, etc.).

TVA Reply to APLA-34/36

The Seismic Individual Plant Examination of External Events (IPEEE) Reports for Unit 1 and Units 2/3 were provided in submittals dated January 14, 2005 (ML050210092) and June 28, 1996, respectively. As stated in these reports, the Seismic Review Team (SRT) reviews and walkdowns performed on the containment did not reveal any significant vulnerabilities. The HCLPF for the containment is greater than 0.3g, based on SRT reviews, walkdowns, and Appendix A of EPRI Report NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin."

NRC Request APLA-35/37

Discuss whether the fire risk evaluation contained in the IPEEE addresses spurious actuations (hot shorts) of the containment isolation valves or the need for the COP credit. If not, address the change in risk considering COP and submit a summary of the updated results.

TVA Reply to APLA-35/37

As discussed with the NRC staff, the response to this RAI will be submitted separately.

NRC Request APLA-36/38

Discuss when and how the peer review for the Unit 1 PRA was conducted (identify the participants, describe what methods were used, etc.), and submit the peer review report. Indicate which comments may significantly affect the PRA results, insights, and conclusions concerning the proposed EPU. Also, discuss the plans and timetable to resolve comments and revise the PRA model.

TVA Reply to APLA-36/38

The peer review of the Unit 1 PRA was conducted in September/October 2006. The peer review team used Revision A-3 NEI draft "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance", NEI 00-02, dated June 2, 2000 as the basis for the process to conduct the review. However, the technical requirements criteria for the review are derived from the current ASME PRA Standard Addendum B. In addition, NEI 05-04, "Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard", January 2005 provides additional guidance for performing follow-on PRA peer reviews.

The PRA peer review team was composed of contractor personnel knowledgeable in PRA issues and experienced in the performance and application of PRAs. The PRA peer review team also included peers who are knowledgeable in PRAs for plants similar to BFN.

Table APLA-36/38.1 lists the Level A and B Facts and Observations (F&O) that were identified during the peer review. Each F&O has been reviewed to evaluate its potential impact on the PRA model. The F&Os which were expected to affect core damage frequency (CDF) were incorporated into the model. The F&Os incorporated into the model and the effect on the BFN Unit 1 CDF of 1.77E-06 that was previously provided in the reply to RAI SPLB-B.7 in our December 19, 2005 submittal (ML053560194) are discussed below.

- Potential common cause miscalibration of the reactor pressure vessel (RPV) low pressure interlock sensors for LPCI and CS injection. This was identified under several F&Os, including SY-B11-1, HR-C3-1, and QU-D1a-4. Incorporation of this change results in an increase in CDF of approximately 1.63E-06.

- The updated base model also includes consideration for the operations team, including the Technical Support Center (TSC) and oncoming shifts, to align decay heat removal. This impact was separately questioned (and the basis justified) during the independent review performed for the Level A F&O. This issue was also identified under F&O HR-G7-2. Incorporation of this change results in an increase in CDF of approximately 6.5E-07.

- Incorporation of the remaining F&Os that were considered to be potentially significant results in an increase in CDF of approximately 6.16E-07. These items included:
  - Comments from the independent model review performed to resolve the Level A F&O. This also incorporated several F&Os, such as QU-A2a-2 concerning unusually low conditional CDF values for several initiating events and LE-E4-1 concerning LERF contributions for various initiating events.
  - Modeling of ATWS, including frequency (see IE-B3-3) and success criteria (see SC-B1-6 and QU-D1a-2).
  - Modeling of break outside containment, including separation into two new initiating events (feedwater and main steamline break – see EI-C3-1) and reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) isolation on high steam tunnel temperature (identified under F&O SY-B2-1 and SY-B14-3), as well as general modeling of impacts identified in QU-D1a-1.

The Level A and B F&Os are expected to have a similar impact on both pre-EPU base models as well as the current model that includes EPU. The effect on the EPU and COP delta CDF values, if any, is expected to be much smaller than the impact on the base CDF given above. Therefore, they would not significantly affect the PRA results, insights, and conclusions concerning BFN EPU.

TVA is currently engaged in upgrading the BFN PRA consistent with the quality requirements of Regulatory Guide 1.200, R1. This project includes implementation of a new PRA software platform (CAFTA), upgraded documentation, and resolution of all pertinent F&Os from the previous peer certification reviews. At the completion of this work, a self-assessment (gap analysis) is planned, followed by a peer certification review and subsequent resolution of all F&Os in the new model. This project is currently targeted for completion in June 2008.

Table APLA-36/38.1 - Level A and B Facts and Observations

F&O	Level	Description
Initiating Event PRA Element IE		
IE-A4-7	B	Manual Shutdowns Manual shutdowns are not included in the model. Some manual shutdowns are from >40% power. (See IE-B3 for additional implications.)
IE-A4-9	B	Screening for various bus failures indicates "Possible reactor scram." Since some of these impacts appear to include other impacts such as loss of some condensate/booster pumps, these should be more fully explained particularly in light of EPU requirements.
IE-A5-1	B	There is no evidence that (a) events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions), and for which it is determined that the event could also occur during at-power operation. (b) events resulting in a controlled shutdown that includes a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to at-power operation.
IE-B3-1	B	Missing AC bus initiators Loss of a bus seems to only result in one initiator type. Other plants have initiators for some of the AC buses because they not only cause an initiator (e.g., IMSIV) but also impacts mitigation systems. Example Loss of 4KV RMOV Board 1A results in IMSIV

F&O	Level	Description
		but also causes loss of power to RHR HX - FCV-23-34, FCV-23-40. Failure of this bus means that it cannot be powered by alternate source. This is more severe than IMSIV. Review other electrical (AC/DC) buses that should be added to the model because they cause an initiating event and consequential failure of mitigating systems.
IE-B3-3	B	<u>ATWS Initiating Events</u> Potential impact on ATWS as a core damage and LERF contributor hinges on ensuring that all of the appropriate initiating events are input to the ATWS event trees. The initiating events that are used to enter the failure to scram event trees should be checked for completeness. The use of LOCHSA as an initiating event appears to be missing some initiating events that should be accounted for. These may include Loss of Plant Control Air (LOPA) and LRCW.
IE-C3-1	B	<u>BOC</u> The derivation of the BOC initiating event does not appear to be documented in the initiating event notebook, although the quantified value is presented. See IE-B4-1.
IE-C10-1	B	No documentation in the IE notebook of COMPARISON of results and EXPLANATION of any differences in the initiating event analysis with generic data sources to provide a reasonableness check of the results.

F&O	Level	Description
IE-C12-1	B	Review of contributing sources of ISLOCA is incomplete. For example, a RWCU control valve malfunction is not considered.
IE-D1-2	B	Initiator values in documentation is different than what is in the model. 1. FLBR1 = 1.20E-3 in the model but Table 7 in the Internal Flooding Notebook says it should be an order of magnitude higher (1.20E-2). 2. Other initiator values in the model are close but not accurate to what is written in the Initiating Event notebook.
IE-D2-2	B	IE notebook discussion regarding calculation of LOSP frequency including Table 4-10 is outdated and incorrect. TVA presented a proposed revision to the summary of the LOSP initiating frequency results. This appears to resolve the discrepancy in the documentation presented to the Team.
IE-D3-1	B	<u>Key Modeling Uncertainties and Key Assumptions</u> There is no discussion of the sources of key modeling uncertainties and key assumptions. This is one of the Supporting Requirements of the ASME PRA Standard.
Accident Sequences PRA Element AS		
AS-A3-1	B	<u>Standby Coolant System</u> The use of RHRSW cross tie to the RPV for RPV injection is limited because of the discharge shutoff head of the RHRSW system. The RHRSW system has a listed shutoff head of 160 psig. For RPV injection the

F&O	Level	Description
		combined: • Elevation head • RPV back pressure • Containment pressure may exceed this shutoff head.
AS-A3-2	B	<u>Vent</u> The hard pipe vent consists of air operated AOV butterfly valves. It is not believed that these can be finely tuned to control containment pressure within a very small pressure band (45-55 psig).
AS-A4-1	B	<u>Loss of DHR with Vent Success</u> The accident sequence evaluation assumes that venting is controlled in a tight band of 45-55 psig. This is assumed to prevent loss of NPSH on the ECCS pumps. This appears optimistic. The ability to vent the containment and to retain sufficient NPSH to operate ECCS pumps is considered to be inadequately evaluated. No calculations are presented to show: (a) The ability to control vent pressure with the hard pipe vent AOVs. (These are believed to be very difficult to achieve a narrow band of pressure control.) (b) The non-condensibles are lost from containment. It is believed that NPSH will continue to decrease as the non-condensibles are vented from the containment.
AS-A9-1	B	It appears that realistic

F&O	Level	Description
		analyses/evaluations that are applicable to the plant and used to support success criteria are based on use of conservative FSAR and generic sources (NUREG/CR 4550, Vol 1.) However, it seems to have been validated by plant specific TH analysis using MAAP or other Codes, but Table 2-4 (Front Line Notebook) does not credit use of either generic or plant specific TH analysis.
AS-A10-1	B	It does not appear that a distinction is made between LOCA and General transients regarding HRA failure probability. Maintaining level during a LOCA or depressurizing the RPV given a LOCA with no high pressure injection should have higher failure Human Error Probabilities (HEPs) than during a general transient where more time is available to take action because of the longer time before reaching top of active fuel.
AS-B6-1	B	The top SBO sequence (Rank 38, 8.3E-9) has the following characteristics: • Weather induced LOOP • CCF of all EDGs • RCIC/HPCI operate for 4 hrs. • No AC recovery at 6 hrs. • Successful HPCI/RCIC level control for 4 hours The sequence appears to assume RPV is depressurized, but the DC power is assumed in the documentation to last 4 hours. Therefore, the RPV would repressurize. No crew action is included to use the alternate PSP curve and avoid RPV depressurization on PSP.
AS-B6-2	B	NPSH impact if Cont. Isol Fails during loss of DHR

F&O	Level	Description
		There are TVA deterministic calculations that indicate that a containment overpressure is required to provide adequate NPSH when torus cooling is ineffective. If containment isolation fails, this overpressure may not exist. Accident sequence evaluation does not have a failure mode that containment isolation fails <u>and</u> there is a loss of torus cooling during a response to an isolation event. This sequence could lead to the loss of non-condensibles and cause inadequate NPSH for the ECCS pumps from the suppression pool <u>and</u> a requirement to depressurize.
AS-B6-3	B	<u>Steam Binding and Loss of NPSH Due to Venting</u> Containment venting may result in the loss of non-condensibles and flashing of fluid at pump suction. The combination of these effects could disable ECCS pumps from the torus.
AS-C3-1	B	<u>Key Modeling Uncertainties and Key Assumptions</u> There is no discussion of the sources of key modeling uncertainties and key assumptions. This is one of the Supporting Requirements of the ASME PRA Standard.
Success Criteria PRA Element SC		
SC-A2-2	B	Need to verify that EPU power level is adequately treated in the success criteria given that generic calculations are being used.
SC-A2-4	B	RCIC capability to respond is said to be successful for SL, SORV, and IORV in the success criteria tables. However, it was

F&O	Level	Description
		noted that: 1. Size for Small LOCA is not defined 2. Water demand for SL, SORV, IORV may be beyond the capacity of RCIC alone
SC-A4a-1	B	IORV success criteria of using PCS and FW considered inappropriate because the crew will close the MSIVs to limit the cooldown rate. This was confirmed by EOP coordinator, Mike Morrow, in the Team's interview with him during the on-site peer review.
SC-A6-1	B	Unable to CONFIRM fully that the bases for the success criteria are consistent with the features, procedures, and Operating philosophy of the plant. Examples of plant specific issues that are to be addressed because of apparent procedural conflicts include: • Assumption that the crew would <u>not</u> close MSIVs given an IORV/SORV • Assumption that the crew would <u>not</u> emergency depressurize the RPV given flooding of the secondary containment • Assumption that the crew would <u>not</u> emergency depressurize when exceeding the PSP under SBO conditions • Assumption that the crew could control containment pressure with venting in a narrow pressure range
SC-B1-1	B	It appears that SC is based on use of conservative FSAR and generic sources (NUREG/CR 4550, Vol 1). Plant specific TH analysis using MAAP or other Codes are available at TVA, but Table 2-4 (Front Line

F&O	Level	Description
		Notebook) does not credit use of either generic or plant specific TH analysis.
SC-B1-4	B	As-Built, As-Operated Plant The Unit 1 PRA is constructed in parallel with the Unit 1 upgrade and EPU implementation. Certain assumptions regarding initial CW, condensate, and FW availability are made. These should be confirmed when plant starts operation. This may affect which events can cause an initiating event. 4kv board 1A, 1B, 1C.
SC-B1-6	B	ATWS Success Criteria Table 2-3/Table 2-4 indicate 1 SLC pump is adequate for reactivity control. HRA indicates 10 min. is available to start SLC. Given these conditions it is judged that depressurization is required based on the HCTL. Provide a calculation that supports the use of HPCI or RCIC for 24 hr. mission time under these conditions. (Note that since FW, CRD may be operating over some portion of the time frame. These injection sources need to be included in the deterministic calculation of power and torus temperature calculation. Examples: Turbine Trip with RPT success should result in normal RPV level control and ~60% power. This means 30% power is going to the torus until RPV level is lowered. At what level is power less than the TBV capacity. What model is used (Chexal Layman or NRC-TRAC). The ATWS success criteria and the

F&O	Level	Description
		implementation in the rules are important because of the impact on CDF and LERF.
SC-B5-2	B	<u>BOC/ISLOCA</u> The treatment of the BOC/ISLOCA initiators in the success criteria and in the event trees is not clear. This treatment of BOC in the model also appears inconsistent with the stated intent of the model during the on-site visit. See the QU element for further discussion of the BOC success criteria and modeling.
SC-C3-1	B	Key assumptions and key sources of uncertainty associated with the development of success criteria are not documented.
Systems Analysis PRA Element SY		
SY-A4-1	B	<u>Confirmation of the System Inputs</u> There is no evidence of a review by system managers, operations, or other site personnel to confirm the inputs provided by the system notebooks particularly the assumptions.
SY-A19-1	B	<u>SRV</u> There is no discussion of the reclosure of the SRVs if the differential pressure between the pneumatic supply and the containment drops too low. This eventually could occur for loss of DHR sequences where venting is unsuccessful.
SY-A20-1	B	Level Control by Steam driven pumps (HPCI & FW) Maintaining RPV Level control is difficult if not impossible since within 15 minutes decay heat requires only approximately 600

F&O	Level	Description
		GPM to maintain level.
SY-B6-1	B	SRV The dependency treatment of SRVs on pneumatic supplies is considered suspect. It appears that the accumulators are asserted to be the basis for 24 hour mission support for SRV operation. This is not documented as to the basis. P.7 Section 1.4.1.4 states that air and power are required support systems and Table 2 provides the list of supports. SRV SN - P. 18 1. Each of the SRVs provided for automatic depressurization is equipped with an air (nitrogen) accumulator and check valve arrangement. These accumulators are provided to ensure that the valves can be held open following failure of the pneumatic supply to the accumulators, and each is sized to contain sufficient air for a minimum of five valve operations. Table 2 only provides the DC support, the adequate pneumatic supplies for 24 hours is not provided. There does not appear to be a split fraction that is dependent on the availability of the pneumatic supply.
SY-B11-1	B	Low Pressure Permissive Confirm the treatment of the miscalibration of the low pressure permissive for the CS and LPCI injection valves. It is believed that the gross miscalibration of the low pressure permissive setpoint for multiple

F&O	Level	Description
		sensors is not included in the model.
SY-B14-1	B	SRVs The dependency treatment of SRVs on pneumatic supplies is considered suspect. It appears that the accumulators are asserted to be the basis for 24 hour mission support for SRV operation. This is not documented as to the basis.
SY-B14-3	B	HPCI and RCIC isolation on high tunnel temperature were identified by operations personnel as being present in the plant. These do not appear to be in the model.
Human Reliability PRA Element HR		
HR-B2-1	B	Supporting requirement HR-B2 states: "DO NOT screen activities that could simultaneously have an impact on multiple trains of a redundant system or diverse systems." HR-C3 requires that miscalibration be included as a mode of failure of initiation of standby systems. For the pre-initiator HEP assessment, the majority of the calibration procedures were screened out and are not included in the quantified model. These impact redundant trains even though they are performed on a staggered basis. See also HR-C3-1.
HR-C3-1	B	Supporting requirement HR-C3 requires inclusion impact of miscalibration as a mode of failure of initiation of standby systems. HR-D5 requires an assessment of the joint probability of those HFEs identified as having some degree of dependency. A number of miscalibrations are discussed

F&O	Level	Description
		in Table 3 4 of the HRA Notebook. Some miscalibrations have HEP-type identifiers (i.e., HAEDG1, HAHPC1) but could not be located in the model. See also HR-B2-1.
HR-G4-2	B	<u>ORE</u> Avoid use of ORE method or verify its applicability and results for the plant specific application. See NUREG-1842 (Draft) comment: An extensive critique of the HCR/ORE method is included in NUREG-1842: • The ability to adequately address the range of plant conditions and PSFs that could bear on performance in an accident scenario (regardless of the approach for obtaining response times) has not been demonstrated. • Guidance for use of expert judgment to obtain estimates of crew response times is not provided. (This creates an issue of validity and reliability). • The validity of generalizing simulator results from ORE experiments to plant-specific analyses was not demonstrated. • The method does not provide a systematic approach to identify important aspects of human performance for the actions modeled in the PRA (an important goal of the HRA). • Until the suitability of using the standard normal distribution is demonstrated and the method is implemented through an adequate

F&O	Level	Description
		number of plant-specific simulator runs to obtain the relevant model parameters, use of the HCR/ORE TRC is not appropriate for regulatory applications. • Because of these limitations, it is uncertain that using this method will yield appropriate "relative values" of HEPs and, hence, appropriate safety insights and improvements.
HR-G4-3	B	There is no discrimination in the time dependency of the RPV depressurization HEP as a function of the initiator and sequence: • Small water LOCA • Small steam LOCA • Medium steam LOCA • Medium water LOCA • IORV • Transient • ATWS All use the same HEPs.
HR-G6-3	B	A 100% success HEP at preserving the main condenser as a heat sink under IORV/SORV conditions is assumed in the model. The IORV success criterion of using PCS and FW is considered inappropriate because the crew will close the MSIVs to limit the cooldown rate. This was confirmed by EOP coordinator Mike Morrow in the Team's interview with him during the on-site peer review.
HR-G7-2	B	HRA values below 1.0E-4 are suspect. The EPRI HRA Calculator defaults to 1.0E-4. Also a note to the HR High Level Requirements provides for a review of those

F&O	Level	Description
		HEPs computed to be less than 1.0E-4. HPSPC1 = 6.10E-6 “Align RHR for Suppression Pool Cooling” and does not meet the recommended minimum. The HEP for suppression pool cooling initiation at 2 hours may be higher than 6E-6. The initiation of suppression pool cooling later in the sequence of events is not asked by the model. At longer time with additional crew and TSC manned, the HEP could be as low as 1E-6. The model does not currently address the time dependence of its HEP in a comprehensive fashion.
HR-I2-1	B	Supporting requirement HR-I2 provides documentation requirements for Element HR. There is not a comprehensive list of the HFEs used in the model in the HR Notebook. There are several HFEs used in the model, but missing from the notebook; Example: OP1 and OP2 used in internal flooding.
HR-I3-1	B	Supporting requirement HR-I3 requires documentation of the key assumptions and key sources of uncertainty associated with the human reliability analysis. There is no discussion in the HRA Notebook of key assumptions or sources of uncertainty.
Data Analysis PRA Element DA		
DA-C4-1	B	Supporting requirement DA C4 calls for developing a clear basis for the identification of events as failures and distinguishing between those degraded states for which a failure, as modeled in the PRA, would have occurred during the mission and those for which a failure would

F&O	Level	Description
		not have occurred. The DA Notebook does not contain the criteria that was used for identifying events as failures.
DA-C5-1	B	Supporting requirement DA-C5 describes how repeated plant-specific component failures occurring within a short time interval should be counted (as a single failure if there is a single, repetitive problem that causes the failures and as only one demand). The DA Notebook does not contain any information on how repeated plant specific component failures were counted.
DA-C8-2	B	The plant specific operational records to determine the time components were in standby were not reviewed. Standby components for running systems should be evaluated. This includes times in standby due to seasonal effects (e.g., CW, FW, condensate, EECW, RCW pumps).
DA-C13-1	B	Supporting requirement DA-C13 calls for an examination of coincident unavailability due to maintenance for redundant equipment (both intra- and inter-system) based on actual plant experience. No coincident events were found in the model. There was no discussion in the DA Notebook that this examination was performed.
DA-E3-1	B	Supporting requirement DA-E3 requires documentation of the key assumptions and key sources of uncertainty associated with the data analysis. There is no discussion in the DA Notebook of key assumptions or sources of

F&O	Level	Description
		uncertainty.
Internal Flooding PRA Element IF		
IF-A3-1	B	Flood Sources There are some flood sources that appear to not be developed. The two principal ones of note are the RHRSW supply to the RHR Hx in the corner room and FPS stations in the Reactor Building. RHRSW pipe rupture, valve rupture, or Hx rupture could cause massive flooding in the RB. This consequence may occur as a result of an initiating event that results in a "shock" to the RHRSW system. Crew response to the flood needs to be assessed to ensure it is adequately modeled.
IF-C2a-2	B	A time of 20-30 minutes is assigned to operator action OP1. The basis of this timing is not adequately discussed. No other flooding related timing is noted. This F&O may also affect SR C3c, C8, and E5a.
IF-C2c-4	B	EECW Failure (Flood) in Reactor Building – Large Flood The top flood sequence credits the main condenser as a heat sink and FW as an injection source. However, the RB flood would require emergency depressurization. Therefore, it appears incorrect to credit the main condenser as available.
IF-C2c-6	B	Derivation EECW/RHRSW Rupture in RB (FLRB2) Flood frequency = 6.5E-4/Rx Yr Fail to isolate = 3E-3

F&O	Level	Description
		The fail to isolate appears optimistic and is not documented in the material presented to the Peer Review Team.
IF-C2c-7	B	There is no discussion of the evaluation of overspray by fire protection piping or other sources. Typically, this focuses on critical powerboards as these can impact many components. This may also apply to SR IF-B3, IF-C3.
IF-F3-1	B	There is no specific discussion of key assumptions and sources of uncertainty as noted by ASME requirements.
Quantification PRA Element QU		
QU-A1-1	B	Manual shutdowns not included in PRA model. This reflects on the possible lack of review of the final model. (This is a duplicate of F&Os that appear in the IE element.)
QU-A1-3	B	The derivation of the ATWS initiator frequencies are not defined in the model or documentation presented to the team. The response presented during the week was: LOCHSA = (IMSIV + LVC1 + TBU1) LOFWA = (TLFF + PLFW1 + TCLF + PLCF) LOOPA = (IEGRLP + IEPCLP + IESWLP + IESWLP) However, loss of plant air and loss of RCW is not included in this list of ATWS challenges.
QU-A2a-2	B	Provide an analysis of the CCDP to see if there are unusually low CCDP for certain initiators. For example, the IORV (IOOV)

<table>
<tr><th>F&O</th><th>Level</th><th>Description</th></tr>
<tr><td></td><td></td><td>initiator has a CCDP = 3.8E-7.

TMSIV has CCDP = see below

CCDP for IOOV should be very similar between the two. (See ISCRAM for example.)

Is there an assumption that MSIVs remain open and PCS is available for the IOOV initiator?

<table>
<tr><th></th><th></th><th>IE</th><th>CDF</th><th>CCDP</th></tr>
<tr><td>IOOV</td><td>=</td><td>1.3E-2</td><td>5.2E-9</td><td>3.8E-7</td></tr>
<tr><td>IMSIV</td><td>=</td><td>5.6E-2</td><td>1.1E-7</td><td>2.0E-6</td></tr>
<tr><td>ISCRAM</td><td>=</td><td>8.6E-2</td><td>3.8E-8</td><td>4.4E-7</td></tr>
<tr><td>LRCW</td><td>=</td><td>7.95E-3</td><td>8.6E-8</td><td>1.1E-5 (Loss of CRD)</td></tr>
</table>

The crew generally will close the MSIVs given an IORV to reduce the cooldown rate. Therefore, crediting the PCS for IORV would seem to be inappropriate. This was confirmed with Mike Morrow (EOP coordinator) during the peer visit.</td></tr>
<tr><td>QU-A4-1</td><td>B</td><td><u>Recovery</u>

The justification for the recovery of the PCS depending on the initiating event is considered questionable. The recovery action is strongly dependent on hardware availability, dependent crew actions, and functional effects of the accident sequence such as required emergency depressurization. None of these influences is treated in detail in the application of the PCS recovery terms that are used in the model. These effects influence all types of initiation including:

• MSIV Closure
• Loss of Condenser Vacuum
• Flooding initiators</td></tr>
</table>

<table>
<tr><th>F&O</th><th>Level</th><th>Description</th></tr>
<tr><td></td><td></td><td>There are “recoveries” that are introduced into the PRA for reopening the MSIVs to restore FW or PCS. They include such variables as:

<table>
<tr><th>Variable</th><th>Value</th><th>Description</th></tr>
<tr><td>PCSR3</td><td>2.86E-2</td><td>Restore PCS given MSIV closure</td></tr>
<tr><td>PCSR4</td><td>0.126</td><td>Restore PCS given Loss of PC Air</td></tr>
</table>

Some of these recoveries are documented in App C of the Data Notebook, but are:

(a) Not plant specific
(b) Are based upon assumptions that have little or no basis

This seems to conflict with the intent of the ASME PRA Standard that requires either plant specific data or an HRA evaluation that is plant specific.

Note that no documentation of PCSR4 (restore PCS given loss of plant control air) has been found.</td></tr>
<tr><td>QU-A4-2</td><td>B</td><td>The model credits a number of recovery actions that involve the PCS. These non-recovery probabilities are noted to be small and further justification seems to be required. Specifically, EOPs require RPV depressurization for various reasons including: PSP, RPV level, containment pressure, torus level, etc. It is not clear that the recoveries modeled could be effective under such conditions.</td></tr>
<tr><td>QU-D1a-1</td><td>B</td><td><u>BOC</u>

Model does not appear to produce sequences with BOC * Fail to Isolate (6.8E-4 * 1E-4 ~ 6.8E-8/yr). This indicates that these</td></tr>
</table>

F&O	Level	Description
		sequences are either crediting additional undocumented mitigation or they are not handled correctly. It is believed that extensive adverse impact on mitigation systems in the R.B. may result however, the extent of this impact does not appear to be documented and the model may assume no impact. The Support System notebook indicates, on Page 3-4, that failure to isolate a BOC initiator would lead to failure of equipment in the Reactor Building. This seems to be a reasonable modeling decision considering the potential for significant environmental impacts. However, no sequences related to this modeling could be found. Event tree rule modeling was reviewed and the only related impacts could be found for RCIC and HPCI. It would appear that there has been a modeling oversight for top events related to Core Spray, LPCI, etc.
QU-D1a-2	B	<u>ATWS</u> The ATWS accident sequence quantification (rules) appears to send sequences to core damage if only HPCI fails given a failure to scram. This does not appear to allow the use of depressurization and low pressure makeup as a successful mitigation path. This appears to be overly conservative. No documentation of this treatment could be found.
QU-D1a-3	B	<u>FLTB2</u> The small Turbine Building flood is described briefly in the Internal Flood Notebook as causing failure of the FW and

F&O	Level	Description
		condensate system. (See Section 4.3) However, no discussion of the status of the main condenser under the conditions imposed by this flood is provided. The PRA model does not include a failure of the main condenser function as a result of the FLTB2 flood. This appears to conflict with the intent of the documentation. (See also QU F&O D1A.)
QU-D1a-4	B	The Support System event tree SIGL models the low RPV pressure permissive signals for ECCS (Top Events NPI, NPII). However, the system models for core spray, LPCI, etc. appear to allow injection if the low RPV level signal OR the permissive is successful (e.g., MACRO SGI:= LV=S+DW=S*NPI=S). Other logic rules and MACROs seem to have a similar treatment. This modeling would be true for pump starts but the injection valves require the low RPV pressure signal to open. This requirement was not found modeled nor was some related operator intervention found in the model.
QU-D1a-7	A	<u>General Model Review</u> The PRA model was reviewed by the PRA Team during the on-site review. This review compared the model with the documentation and the oral descriptions of the intent of the model. A number of modeling anomalies were found (see related QU-D1a Supporting Requirement F&Os). Since the PRA Peer Review Team only reviewed a sample of sequences, it is recommended that a more rigorous sequence review be performed in conjunction with Event Tree structure and rule review.

F&O	Level	Description
		See the attached Differing Professional Opinion from Mr. Omer Gokcek. Labeled QU-D1a-7a.
QU-E1-1	B	Key Model Uncertainties Section 3.5 addresses key sources of uncertainty in addition to uncertainty related to the statistical treatment of data. However, this list is fairly limited. The assessment of the key model uncertainties appear to be of limited scope. EPRI has developed a methodology for the disposition of model uncertainties and for their quantitative evaluation using sensitivity studies. See EPRI TR-1009652 dated December 2004.
QU-F1-2	B	The quantification of the model was modified from that in the documentation. The model and the associated documentation need to be made consistent. For example, on page A-109 in the Event Tree notebook, rule "VNTF" is "OWWV=F". In the model, its "OWWV=F +-(CS=S+INJI=S + INJII=S + LPCII=S*XTV=S*U2X=S)". Another example is the discussion of Top Event TOR for HPCI in the HPRCET event tree (page 5-31 of ET Notebook).
QU-F4-1	B	Section 3.5 addresses key assumptions. However, this list is fairly limited
QU-F5-1	B	There is no discussion of the degree to which any limitations of the model may impact applications as required by ASME requirement QU-F5.
LERF Analysis PRA Element LE		
LE-E4-1	B	The contributions to LERF generally appear

F&O	Level	Description
		reasonable given the current Level 1 results. (There are comments on the Level 1 results that may propagate to the Level 2 results.) However, the LERF assessment appears to take too much credit for AC power recovery for the following contributions to LOOP initiator: • weather 10% non-recovery (conditional LERF given CDF) • grid 9% non-recovery (conditional LERF given CDF) This should be investigated. ATWS contribution may be disproportionately high.
LE-E4-2	B	During the on-site review, TVA provided the following information: The Level 2 model fault trees are CAFTA fault trees (pictures) that were developed in the time frame 2000-2001. Point estimates are provided in the L2/LERF documentation. Efforts by TVA/ABS to reproduce the fault tree values using the information provided are unsuccessful.
LE-F2-1	B	Key sources of uncertainty and other limitations are not identified. Applicable to LE-G4 and LE-G5 also.
LE-G1-1	B	The quantification of the model was modified from that in the documentation. The model and the associated documentation need to be made consistent.
LE-G4-1	B	Key sources of uncertainty and other limitations are not identified.

F&O	Level	Description
		Applicable to LE-G5 also.
LE-G5-1	B	Key sources of uncertainty and other limitations are not identified. Applicable to LE-G4 also.
Maintenance and Update PRA Element MU		
MU-A2-1	B	Probabilistic Safety Assessment (PSA) Program, SPP-9.11 Rev. 0000 does not currently address to include monitoring of changes in PRA technology and industry experience that could change the results of the PRA.
MU-B4-1	B	Probabilistic Safety Assessment (PSA) Program, SPP-9.11 Rev. 0000 does not currently address that, "PRA Upgrades shall receive a peer review (in accordance with the requirements specified in Section 6 of the ASME PRA Standard) for those aspects of the PRA that have been upgraded. Refer to Section 2 of the ASME PRA Standard for the distinction of a PRA Upgrade versus PRA maintenance and update.
MU-C1-1	B	Probabilistic Safety Assessment (PSA) Program, SPP-9.11 Rev. 0000 does not currently have a process to ensure, " that the cumulative impact of pending changes is considered when applying the PRA"
MU-D1-1	B	Probabilistic Safety Assessment (PSA) Program, SPP-9.11 Rev. 0000, Section 3.2.2, Update of Existing Applications, states that Update of existing PSA supported applications will be evaluated with each PSA MOR issuance per NEDP-26. There is no discussion of the process.

NRC Request APLA-37/39

Explain how COP credit maintains the defense-in-depth concept provided Section 2.2.1.1 of Regulatory guide 1.174. Specifically address each of the seven bulleted items with particular emphasis on the fifth item (independence of barriers is not degraded).

TVA Reply to APLA-37/39

Regulatory Guide 1.174 section 2.2.1.1 provides seven elements used to address defense in depth. Credit for COP to ensure ECCS pump function in analyses licensing basis events as opposed to neglecting it as a conservative assumption was evaluated under these elements.

- *A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.*

  The requested BFN EPU licensing basis change does not alter the basic boiling water reactor design currently employed on the three BFN units. This basic design has a well-accepted defense-in-depth balance. COP credit is required in the analyses of certain deterministic scenarios for the BFN design; however, due to the unlikely nature of these scenarios, the fundamental balance between core damage prevention, containment failure prevention, and consequence mitigation is not significantly impacted.

- *Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.*

  The existence of adequate COP to ensure ECCS NPSH results from the thermodynamic response to the containment to the temperature in the suppression pool water and the pressure integrity of the containment. These are principle design features and there is no additional reliance on programmatic activities. An additional operator action was needed in the Appendix R event which is similar to other operator actions and, therefore, does not result in over-reliance on programmatic activities.

- *System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).*

  Risk evaluations were performed to bound the change in core damage frequency considering ECCS dependency on COP compared with the assumption that no such dependency exists. These results were provided in TVA submittals dated July 21, 2006 (ML062090071) and September 15, 2006 and demonstrate that reliance on COP does not create a risk outlier.

- *Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.*

  In the BWR design, low pressure ECCS relies on the containment as its water source. The containment in turn relies on the containment cooling mode of the RHR ECCS system for heat removal and ultimately containment integrity. Consideration of COP for determining ECCS NPSH adequacy constitutes an additional dependence in which ECCS relies on the pressure boundary integrity of the containment. These dependancies between two safety related plant features establish the containment as a support feature for ECCS and do not introduce a new common mode failure.

- *Independence of barriers is not degraded.*

  COP constitutes a dependancy between fuel cladding barrier and the primary containment barrier only in low probability event scenarios as shown by risk evalautions provided in TVA submittals dated July 21, 2006 (ML062090071) and September 15, 2006 . Therefore, an appropriate level of independence is maintained and independence is not degraded.

- *Defenses against human errors are preserved.*

  The EPU change request in general, and the requested COP credit in particular, do not require significant changes to the duties of the plant operations staff. No significant additional human error situations are added via this change request, and the existing human error prevention tools will remain effective in the proposed EPU operating environment.

- *The intent of the General Design Criteria in Appendix A to 10 CFR Part 50 is maintained.*

  See Appendix A to the BFN Updated Final Safety Analysis Report (UFSAR) for an extensive discussion on BFN's compliance with the General Design Criteria (GDC) as they existed at the time of BFN initial licensing. Appendix A to 10 CFR 50 currently contains the GDC. A complete listing of these is tabulated below for ease of reference. Those criteria which are impacted by the use of COP credit are addressed. Those GDC on which COP credit is judged to have no or negligible impact are indicated by shading.

I.	***OVERALL REQUIREMENTS:***	
	1.	Quality Standards and Records
	2.	Design Bases for Protection Against Natural Phenomena
	3.	Fire Protection
	4.	Environmental and Dynamic Effects Design Bases
	5.	Sharing of Structures, Systems, and Components
II.	***PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS***	
	10.	Reactor Design
	11.	Reactor inherent Protection
	12.	Suppression of Reactor Power Oscillations
	13.	Instrumentation and Control Instrumentation is available to observe parameters affecting RHR and CS pump NPSH and to allow for proper control of these pumps
	14.	Reactor Coolant Pressure Boundary
	15.	Reactor Coolant System Design
	16.	Containment Design The containment pressure and temperature utilized during events where COP is credited do not approach containment design limits.
	17.	Electric Power Systems
	18.	Inspection and Testing of Electric Power Systems
	19.	Control Room
III.	***PROTECTION AND REACTIVITY CONTROL SYSTEMS***	
	20.	Protection System Functions

	21.	Protection System Reliability and Testability
	22.	Protection System Independence
	23.	Protection System Failure Modes
	24.	Separation of Protection and Control Systems
	25.	Protection System Requirements for Reactivity Control Malfunctions
	26.	Reactivity Control System Redundancy and Capability
	27.	Combined Reactivity Control Systems Capability
	28.	Reactivity Limits
	29.	Protection Against Anticipated Operational Occurrences
IV.	***FLUID SYSTEMS***	
	30.	Quality of Reactor Coolant Pressure Boundary
	31.	Fracture Prevention of Reactor Coolant Pressure Boundary
	32.	Inspection of Reactor Coolant Pressure Boundary
	33.	Reactor Coolant Makeup
	34.	Residual Heat Removal
	35.	Emergency Core Cooling
	36.	Inspection of Emergency Core Cooling System
	37.	Testing of Emergency Core Cooling System
	38.	Containment Heat Removal The containment pressure and temperature utilized during events where COP is credited do not approach containment design limits. The capability to remove heat from the primary containment is retained.
	39.	Inspection of Containment Heat Removal System
	40.	Testing of Containment Heat Removal System
	41.	Containment Atmosphere Cleanup

	42.	Inspection of Containment Atmosphere Cleanup Systems
	43.	Testing of Containment Atmosphere Cleanup Systems
	44.	Cooling Water The calculations which indicate a need for COP credit exists utilize conservative cooling water temperature assumptions. Cooling water temperatures beyond those utilized in these calculations are not expected to ever be experienced, and TS limitations do not permit operation with these temperatures even should they occur.
	45.	Inspection of Cooling Water System
	46.	Testing of Cooling Water System
V.	***REACTOR CONTAINMENT***	
	50.	Containment Design Basis The containment pressure and temperature utilized during events where COP is credited do not approach containment design limits.
	51.	Fracture Prevention of Containment Pressure Boundary
	52.	Capability for Containment Leakage Rate Testing
	53.	Provisions for Containment Testing and Inspection
	54.	Systems Penetrating Containment
	55.	Reactor Coolant Pressure Boundary Penetrating Containment
	56.	Primary Containment Isolation The EPU licensing basis change has no negative impact on the capability or performance of the primary containment isolation valves.
	57.	Closed Systems Isolation Valves
VI.	***FUEL AND RADIOACTIVITY CONTROL***	
	60.	Control of Releases of Radioactive Materials to the Environment

61.	Fuel Storage and Handling and Radioactivity Control
62.	Prevention of Criticality in Fuel Storage and Handling
63.	Monitoring Fuel and Waste Storage
64.	Monitoring Radioactivity Releases