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Constellation Energy
Generation Group

July 17, 2007

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Request for Additional Information Regarding Implementation for Alternative Source Term

REFERENCES:

(a) Letter from Mr. B. S. Montgomery (CCNPP) to Document Control Desk, dated November 3, 2005, License Amendment Request: Revision to Accident Source Term and Associated Technical Specifications

(b) Letter from Mr. P. D. Milano (NRC) to Mr. J. A. Spina (CCNPP), dated December 22, 2006, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 -- Request for Additional Information Regarding Implementation of Alternative Source Term (TAC Nos. MC8845 and MC8846)

(c) Letter from Mr. J. A. Spina (CCNPP) to Document Control Desk (NRC), dated March 22, 2007, Request for Additional Information Regarding Implementation for Alternative Source Term (TAC Nos. MC8845 and MC8846)

Reference (a) submitted a request to revise the accident source term in the design basis radiological consequence analyses. Reference (b) requested additional information needed to complete the Nuclear Regulatory Commission (NRC) review of the proposed change. Our response to Reference (b) is contained in Reference (c). Subsequent to the submittal of Reference (c), conference calls were held with the NRC staff to discuss some aspects of our response. It was determined that additional clarification was needed. The clarifications are contained in Attachment (1). When providing these clarifications, it was determined that several of the marked-up Technical Specification pages provided in References (a) and (c) need to be revised. These revised Technical Specification pages are contained in Attachment (2) and replace the same pages or sections in References (a) and (c).

Also, during internal reviews, it was determined that one of the results provided in Reference (c) was not reported correctly. In the response to Question 19, the maximum hypothetical accident exclusion area boundary dose for the worst-case penetration room pathway was reported as 0.1254 Rem. The correct value is 0.1554 Rem. This change results in a change in the total maximum hypothetical accident exclusion area boundary dose to 1.8529 Rem instead of the reported 1.8229 Rem. The revised exclusion area boundary dose is still well within the regulatory limit of 25 Rem. The revised pages of Reference (c) are contained in Attachment (3).

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Note that this additional information does not change the No Significant Hazards Consideration or the Environmental Consideration discussion contained in Reference (a).

Should you have questions regarding this matter, please contact Mr. Jay S. Gaines at (410) 495-5219.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, James A. Spina, being duly sworn, state that I am Vice President - Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of St. Mary's, this 17th day of July, 2007.

WITNESS my Hand and Notarial Seal:

Janet L. Scally
Notary Public

My Commission Expires: March 1, 2011
Date

JAS/PSF/bjd

Attachments:
(1) Additional Information
(2) Marked-up Technical Specification Pages
(3) Corrected Pages from Calvert Cliffs Nuclear Power Plant Letter dated March 22, 2007

cc: D. V. Pickett, NRC
S. J. Collins, NRC
Resident Inspector, NRC
R. I. McLean, DNR

ATTACHMENT (1)

ADDITIONAL INFORMATION

Subsequent to the submittal of our March 22, 2007 response to a Request for Additional Information, conference calls were held with the NRC staff to discuss some aspects of our response. It was determined that additional clarification was needed. The clarifications are contained below. The numbered clarifications below match the numbered responses in the March 22, 2007 letter.

34. We confirm that leakage of radioactive liquid into the ECCS pump room following an accident is not part of our licensing basis. Therefore, we do not consider a release from the ECCS pump room when performing our accident dose analysis. The ECCS pump room ventilation system is not used or credited in the accident analysis for the control of radioactive material following an accident. Therefore, the radiation monitoring requirements of GDC 64 (draft GDC 17), Monitoring Radioactive Releases, is not applicable to the ECCS pump room. [Note that GDC 60 (draft GDC 70), Control of Releases of Radioactive Material to the Environment, applies to normal reactor operation including anticipated operational transients, but not postulated accidents.] However, as requested, the specific GDC are discussed below.

Although we have requested the removal of the ECCS Pump Room Exhaust Filtration System from the Technical Specifications, the system is described in the Updated Final Safety Analysis Report (UFSAR). Any changes that may be made to system design or operation would be evaluated in accordance with 10 CFR 50.59 to determine if the proposed changes require prior Nuclear Regulatory Commission (NRC) staff approval. Changes to the system would also need to be reviewed to ensure that we continue to meet the GDC as described below. Note that if there were a radioactive release into the Auxiliary Building, the released material would be directed to the Auxiliary Building vent stack, which is a filtered (HEPA filter), monitored release point. Although the ECCS ventilation system does not perform an accident mitigation function, it can be used for other non-safety related functions (such as atmosphere cleanup during refueling outage system maintenance) and we plan to continue to maintain and use the system accordingly.

Draft Criterion 70 – Control of Releases of Radioactivity to the Environment (Category B) The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR Part 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR Part 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Calvert Cliffs Compliance

The wording of the draft GDC 70 refers to the operation of the radioactive Waste Processing System (WPS). Our response to the draft GDC (FSAR Amendment 11) and the final GDC (GDC 60) refers specifically to radioactive WPSs. The proposed changes have no impact on how the requirements of draft GDC 70 are met. As stated in our response to draft GDC 70, the radioactive WPS collects, segregates, processes, and disposes of radioactive solids, liquids, and gases in such a manner as to comply with 10 CFR Part 20. Solid wastes are processed in a batch manner for off-site disposal. Processed liquid wastes and gaseous wastes released to the environment are monitored and discharged with suitable dilution to assure tolerable activity levels on the site and at the site boundary. Holdup capacity in the reactor coolant WPS is 360,000 gallons; the miscellaneous WPS has a storage capacity of 8,000 gallons. All liquid wastes are sampled to establish their acceptability for release. The contents of the waste gas

decay tanks are sampled, and a release rate established consistent with the prevailing environmental conditions. A capability is provided for 60-day holdup of waste gas. In-line monitoring provides a continuous check on the release of activity. Under accident, conditions, radioactive gaseous effluents that may be released into enclosed areas are collected by the ventilation systems and discharged to the plant vent. Permanently installed area radiation detectors and the plant vent radioactivity detectors are used to monitor the discharge levels to the environment. In addition, portable radiation monitors are available on site for supplemental surveys.

Draft Criterion 17 – Monitoring Radioactivity Releases (Category B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Calvert Cliffs Compliance

Under accident conditions, radioactive gaseous effluents that may be released into enclosed areas are collected by the ventilation systems and discharged to the plant vent. Permanently installed area radiation detectors and the plant vent radioactivity detectors are used to monitor the discharge levels to the environment. In addition, portable radiation monitors are available on site for supplemental surveys. Means are also provided for monitoring the containment atmosphere, which include continuous air monitoring. High radiation inside Containment will also cause containment isolation.

Additional Information Request

Following a conference call, the NRC staff requested the following information:

The staff notes that CCNPP does not have a Technical Specification limit on ECCS leakage and affected design basis accident analyses effectively assume no leakage from systems in the ECCS Pump Room. However, it concerns the staff that this assumption is not supported by any test data or reasonable engineering judgment. So, as it is the CCNPP historical design basis, this assumption may suffice for the purpose of radiological consequences in design basis accident analyses, but it does not necessarily provide reasonable assurance that the assumption is adequate to represent actual leakage in the ECCS Pump Room.

A. Therefore, provide a discussion of the means that exist to limit the leakage in the ECCS Pump Room to a value that assures that dose limits will not be exceeded without the HVAC system you propose to delete from the Technical Specification;

Response

Following the TMI accident in 1979, action items were established for the industry to address. One of the TMI Action Items was Lessons Learned action 2.1.6a (also known as III.D.1.1). This item required actions to minimize the release of radioactivity from systems outside Containment. These actions were primarily directed towards minimizing leakage from these systems. Our response to this action item included the addition of a program to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low a practicable to our Technical Specifications. That program continues to be controlled by our existing Technical Specification 5.5.2, "Primary Coolant Sources Outside Containment." The fluid systems located in the ECCS Pump Room are included in the program. This program also contains integrated leak test requirements for the systems of concern. These systems are required to be leak tested every 24 months. The leak test consists of filling the system with water and operating a system pump for

10 minutes. The system is then inspected for leaks using VT-2 visual examination criteria. We reviewed the leak tests performed from 1997 to 2005 to determine the type and quantity of leakage from the safety injection systems found during these tests. The results are given below.

Test Year	Unit 1 Results	Unit 2 Results
2005	No active leakage	No test results located*
2003	LPSI leak-off line plug leak – 2 drops/min	No active leakage
2001	HPSI mechanical seal leak – no leak rate recorded	HPSI threaded connection – 6 drops/min
1999	No active leakage	No active leakage
1997	No active leakage	Containment spray mechanical seal – 20 drops/min

* Documentation of the 2005 test for Unit 2 could not be located. This issue has been entered into our corrective action program.

The high pressure safety injection (HPSI) pumps, low pressure safety injection (LPSI) pumps and the containment spray pumps are all equipped with mechanical seals. The seals were originally supplied by Borg Warner and Durmetalic. The same seals and seal parts are now supplied by Flowserv. The pumps are run quarterly in the recirculation flow line-up for various surveillance tests. During refueling outages, the pumps are tested at design flows. The mechanical seals are the most likely source of pump leakage during pump operation. In-plant observation of the mechanical seals during testing (both quarterly testing and testing in accordance with Technical Specification 5.5.2) shows typical performance of the seals to be 2-to-4 drops per minute leakage during pump startup, followed by non-detectable leakage until the pumps are secured. This data is not formally recorded as part of the required in-service pump testing or surveillance testing. Vendor information concerning seal leakage was not found.

Leakage in plant systems is an issue that receives a great deal of attention. Operators, maintenance personnel, radiation safety personnel and security officers are all trained to look for leakage from plant systems as part of their daily rounds and activities. When a new leak is identified, it is entered into our corrective action program and evaluated in accordance with our fluid leak management program. The fluid leak management program was developed to ensure external leakage from plant systems is classified, prioritized and managed or corrected in a timely manner. This program provides a proactive approach toward preventing leaks. The goal is to have a zero tolerance of detrimental external leakage. The program establishes action levels for leaks depending on the system they are located in and their size. The safety injection systems in the ECCS pump rooms are classified at the highest level of leak tightness. This means that greater than detectable leaks in the safety injection systems will be fixed as a priority 1, 2 or 3 maintenance item depending on the size of the leak and the location. Many leaks are able to be fixed within a very short period of time because they are so minor. Part of the leak evaluation is consideration of the leak on post-accident accessibility or operation of vital equipment. Barely detectable leaks are scheduled to be fixed as part of routine plant maintenance.

B. And discuss whether or not the limiting leakage would require a Technical Specification, if not, why not.

Response

A Technical Specification already exists (TS 5.5.2, "Primary Coolant Sources Outside Containment) to control the leakage of systems outside Containment that could contain highly radioactive fluid.

C. In addition, provide a discussion demonstrating how the dose consequence resulting from the ESF leakage assumed in the submitted design basis LOCA analysis bounds the dose consequence associated with potential leakage from ESF systems located in the ECCS Pump Room. A comparison between the activity mitigating features associated with the analyzed ESF leakage release path and leakage from systems in the ECCS Pump Room should be included in this discussion.

Response

During the recirculation phase following an accident, radioactive sump water is recirculated through the ECCS pumps and could leak through various valves and reach the refueling water tank (RWT). Thus, there is a potential for an unmonitored release pathway through the RWT, which is vented directly to the atmosphere. This source is accounted for in the submitted design basis LOCA analyses. However, due to the implementation of the leakage reduction program in accordance with TMI Action Item III.D.1.1, no leakage is assumed from the ECCS through the ECCS pump room to the Auxiliary Building. However, if there were any fluid leakage into the ECCS Pump Room, any airborne radioactivity in the ECCS Pump Room could be transported to a filtered (HEPA filter), monitored release path (the Auxiliary Building vent stack). This release path is evaluated in the accident dose analysis (but does not assume a release from the ECCS Pump Room). To ensure that any effluent from this release path would have less effect on the Control Room dose than an equivalent effluent from the RWT, we are proposing to modify the ECCS Pump Room ventilation fan control circuit to ensure that the fans do not automatically operate following an accident. When the fan is not operating, gravity dampers in the ventilation ducts close. This eliminates the path from the ECCS Pump Room to the Auxiliary Building vent stack. Without ventilation fan operation, any leakage in the ECCS Pump Room will have no motive force to move it outside of that room. Note that the ECCS Pump Room ventilation system does not provide a cooling function for the equipment in the ECCS Pump Room. The air temperature in the ECCS Pump Room is maintained within limits by the ECCS Pump Room air cooler. Saltwater-cooled fan coil coolers are installed in each ECCS Pump Room to provide room cooling, if necessary, following an accident. These fan coil coolers are contained within the ECCS Pump Room and not connected to the Auxiliary Building ventilation system.

35. To ensure that our Technical Specifications continue to properly reflect the requirements of 10 CFR Part 50, Appendix J, we revise our request to change the maximum allowable containment test leakage rate, L_a, to 0.16 percent of containment air weight per day at P_a. The corrected marked up Technical Specification pages are attached. Please replace the affected pages provided in the March 22, 2007 letter with the attached pages. The analyses provided in the November 3, 2005 and the March 22, 2007 letters accurately reflect a containment leakage rate of 0.16 weight percent per day.

36. Surveillance Requirement 3.7.11.3 requires verification that each spent fuel pool exhaust ventilation system fan can maintain a measurable negative pressure with respect to atmospheric pressure. During operation the spent fuel pool area ventilation system is designed to maintain a slight negative pressure in the spent fuel pool area with respect to adjacent areas, to ensure that all spent fuel pool exhaust is released through the auxiliary building ventilation stack to assure adequate atmospheric dispersion and radiation monitoring. This is consistent with the Calvert Cliffs licensing basis, which only requires that a negative pressure be established. No specific negative pressure was mandated in the licensing basis. The existing Surveillance Requirement was evaluated during the plant conversion to the Improved Technical Specifications and a specific negative pressure and flow rate were determined to be outside of the plant licensing basis. To clarify the Surveillance Requirement, we propose a slight modification to the wording. We propose replacing the wording "with respect to atmospheric pressure" with "with respect to adjacent areas." This will insure that airflow following an accident is into the spent fuel pool area. The appropriate marked up Technical Specification page is attached.

37 and 40. The acceptability of the Control Room operator doses is not dependent upon the closure of the automatic isolation dampers at the Access Control HVAC Unit RTU-1 (AC-11) and Access Control Air Conditioning Unit 13 (AC-13). These dampers provide fresh air to rooms adjacent to the Control Room. Air from these rooms would have to leak into the Control Room via doors or walls.

(a) For the MHA, SGTR, MSLB, SRE, and CEAEE, a loss-of-offsite power (LOOP) is assumed in the accident analyses. A LOOP would stop the fans in AC-11 and AC-13 and would cause the inlet dampers to automatically shut. Even if the safety related dampers both failed open (a double failure), the loss of fans would effectively isolate the Auxiliary Building from the environment, since no driving force or differential pressure would exist to drive air from the environment through the dampers and into the Auxiliary Building. Any air that did leak into the Auxiliary Building through these units would then have to leak into the Control Room via doors or walls.

(b) The fans, dampers, and radiation monitors are safety-related, therefore, for an accident in which no LOOP occurs (MHA, SGTR, MSLB, SRE, CEAEE), the dampers would close on a high radiation signal and the fans would cease operation. Forced ventilation from the west road inlet would dwarf any minor leakage through the access control units. The dampers and radiation monitors would have to experience failures beyond the required single failure to provide a path to the Control Room.

(c) For a FHA, no LOOP is postulated to occur and our design basis does not require the assumption of a single failure for a FHA. Thus the safety related AC-11 and AC-13 dampers are not assumed to fail open.

A sensitivity study was performed to determine the additional margin that could exist if failure of the dampers was assumed in place of the worst case single failure for the accident.

- For an MHA, the additional margin from assuming two trains of Control Room filtration and three trains of containment filtration would dwarf the loss of margin assuming all leakage via the AC-11 and AC-13 pathways. The EAB, LPZ and CR doses would decrease by 0.4993, 0.1203, and 0.6606 Rem, respectively.
- For the CEAEE, SRE, MSLB and SGTR, in addition to the two trains of Control Room ventilation that would be available, the main condensers would also be available and greatly reduce the released activity from that assumed in the analyses of record.

While the dampers are utilized to determine atmospheric dispersion coefficient values, failure of the dampers does not constitute the worst case single failure during a design basis accident. Therefore, the worst case operator doses are not dependent upon the closure of these dampers, and they are not required to be included in the Technical Specifications.

41. To ensure appropriate filtration of the Control Room air, we propose installing an additional HEPA filter unit to the Control Room filtration system described in the November 3, 2005 letter. The new 10,000 cfm filtration system will then consist of two HEPA filtration units in series with a charcoal filtration unit. We currently plan to place one HEPA unit before the charcoal filter and one after the charcoal filter. The addition of one HEPA filter to the planned system requires a change to the CREVS pressure drop criteria shown in Technical Specification 5.5.11.d. We are requesting an increase to the allowed pressure drop across the CREVS filtration system from 4 inwg to 6 inwg to account for the design change from one HEPA in the filtration system to two HEPAs in the filtration system. Technical Specification 5.5.11.d is marked up to show the proposed change.

We propose changing our Ventilation Filter Testing Program test criteria for the charcoal adsorber to ensure charcoal efficiency is correctly credited in the accident analyses. We propose changing the methyl iodide penetration test percentage for the CREVS from 5% to 4.5%. This will support the analyses assumption of 90% efficiency for the charcoal filter in the CREVS as follows. With a 4.5% penetration limit, we then apply a safety factor of 2, resulting in an assumed efficiency of 91% [100% - (4.5%x2) = 91%]. We then also assume a 1% bypass flow, resulting in an overall assumed efficiency of 90% [91%-1%=90%]. The analyses provided in the November 3, 2005 letter and the March 22, 2007 letter conservatively assume a total charcoal filter bank efficiency of 90% for the CREVS. We also propose changing the methyl iodide penetration test percentage for the Penetration Room Exhaust Ventilation System and the Iodine Removal Units from 35% to 34.5%. This change allows the incorporation of a 1% bypass flow assumption, while maintaining the efficiency assumptions used in the accident analyses previously submitted. A marked up page showing the change to the penetration test criteria is attached. Please replace the affected section (TS 5.5.11.c) provided in the March 22, 2007 letter with the attached section (TS 5.5.11.c).

We propose changing the Ventilation Filter Test Program test criteria for the CREVS HEPA filter bank to ensure the CREVS HEPA filter bank is correctly credited in the accident analyses. We propose changing the CREVS HEPA filter bank penetration and bypass test percentage from 1 % to 0.05%. In accordance with Regulatory Guide 1.52, Revision 2, this allows an analysis assumption of 99% for CREVS HEPA filter penetration and bypass efficiency. The analyses provided in the November 3, 2005 letter and the March 22, 2007 letter assume a total CREVS HEPA filter bank efficiency of 99%. A marked up page showing the change to the penetration test criteria is attached. Please replace the affected section (TS 5.5.11.a) provided in the March 22, 2007 letter with the attached section (TS 5.5.11.a).

ATTACHMENT (2)

MARKED-UP TECHNICAL SPECIFICATION PAGES

Calvert Cliffs Nuclear Power Plant, Inc.
July 17, 2007

1.1 Definitions

disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

L_a

0.16%

The maximum allowable containment leakage rate, L_a, shall be ~~0.20%~~ of containment air weight per day at the calculated peak containment pressure (P_a).

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify an OPERABLE SFPEVS train is in operation.	12 hours
SR 3.7.11.2	Perform required SFPEVS filter testing in accordance with the Ventilation Filter Testing Program.	In accordance with the Ventilation Filter Testing Program
SR 3.7.11.3	Verify each SFPEVS fan can maintain a measurable negative pressure with respect to ~~atmospheric pressure~~ adjacent areas.	24 months

5.5 Programs and Manuals

(≤ 0.05% for the CREVS only)

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975, at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System (CREVS)	~~2,000~~ 10,000 cfm
~~Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREFS)~~	~~3,000 cfm~~
Penetration Room Exhaust Ventilation System (PREVS)	2,000 cfm
~~Spent Fuel Pool Exhaust Ventilation System (SFPEVS)~~	~~32,000 cfm~~
IRS	20,000 cfm

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975, at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
CREVS	~~2,000~~ 10,000 cfm
~~ECCS PREFS~~	~~3,000 cfm~~
PREVS	2,000 cfm
~~SFP Ventilation System~~	~~32,000 cfm~~
IRS	20,000 cfm

c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a

5.5 Programs and Manuals

temperature of 30°C and greater than or equal to the relative humidity specified as follows:

ESF Ventilation System	Penetrations	RH
CREVS	4.5 ~~X%~~	70%
~~ECCS PREFS~~	~~50%~~	~~95%~~
PREVS	34.5 ~~35%~~	95%
~~SFP Ventilation System~~	~~15%~~	~~95%~~
IRS	34.5 ~~35%~~	95%

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1975 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Delta P	Flowrate
CREVS	6 ~~X~~ inwg	10,000 ~~2,000~~ cfm
~~ECCS PREFS~~	~~4 inwg~~	~~3,000 cfm~~
PREVS	6 inwg	2,000 cfm
~~SFP Ventilation System~~	~~4 inwg~~	~~32,000 cfm~~
IRS	6 inwg	20,000 cfm

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control for potentially explosive gas mixtures contained in the Waste Gas Holdup System and the quantity of radioactivity contained in gas storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in the ODCM.

The program shall include:

a. The limits for concentrations of oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, including errata, as modified by the following exceptions:

a. Nuclear Energy Institute (NEI) 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after the June 15, 1992 Type A test shall be performed no later than June 14, 2007.

b. Unit 1 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

c. Unit 2 is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, is 49.4 psig. The containment design pressure is 50 psig.

The maximum allowable containment leakage rate, L_a, shall be ~~0.20~~ 0.16 percent of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing, in accordance with this program, the leakage rate acceptance criterion are $\leq 0.60\ L_a$ for Types B and C tests and $\leq 0.75\ L_a$ for Type A tests.

ATTACHMENT (3)

CORRECTED PAGES FROM

CALVERT CLIFFS NUCLEAR POWER PLANT LETTER

DATED MARCH 22, 2007

NUREG/CR-6189 (i.e., the Power's Containment Natural Deposition Model). We used the guidance in NUREG/CP-6189 to reduce the amount of airborne radioactivity in containment after assuming the required release in Reference 2, Appendix H.1. In addition, the CEAE accident release was increased by the radial peaking factor of 1.70, yielding a total release of 42.50%. Therefore, we believe we remain in compliance with the provisions of Reference 2.

Also note that Reference 2 Appendix A.3.2 states that the practice of deterministically assuming that a 50% plateout of iodine from the fuel is no longer allowed since it is inconsistent with the characteristics of the revised source term. Therefore, we believe that the 25% iodine release does not include a deterministically derived plateout. This position is consistent with Reference 2 Appendix H.1.

Maximum Hypothetical Accident (MHA)

1. *In the MHA Design Analysis CA06449, Revision 0, as a design basis, it is assumed that an 8-hour period is required to establish SDC from a HFP condition, which is consistent with "actual plant operation," After the 8-hour period where SDC is established, the postulated RCS activity release to the environment ceases. Because of this dynamic activity release, it cannot be assumed that the activity concentration in the environment, resulting from this release, is at its highest from the onset of the accident-initiated leakage; moreover, and more specifically, the determination of a worst 2-hour EAB dose must account for the actual activity release profile. It appears that the licensee attempted to compensate for this effect by postulating an activity release case that assumes SDC is restored in 2 hours, as opposed to 8. However, this case misses the peak environmental activity concentration associated with the design basis 8-hour release period, which may in fact be even higher than that associated with the hypothetical 2-hour release period case. By assigning the EAB dispersion coefficient from time equal to 0 through the end of the accident, in the 8-hour period case, the code would be allowed to identify the peak 2-hour dose period associated with the DBA release scenario.*

   Please provide verification that the methodology used to determine the worst-case 2-hour EAB dose for the design basis MHA scenario, as described in the submittal, was both conservative and accurate.

Response: Although an MHA release is not assumed to be terminated when SDC is established, we understand that this case may miss the peak environmental activity concentration associated with the design basis release period. By assigning the EAB dispersion coefficient from time equal to 0 through the end of the accident, the RADTRAD code would be allowed to identify the peak 2-hour dose period associated with the DBA release scenario. Therefore, the containment and penetration room pathways were rerun with the EAB dispersion coefficient assigned from 0 to 30 days post-LOCA. Note that the MHA containment and vent stack pathway cases were rerun with a linear ramp isotopic release rate and with containment sprays effective on elemental iodine from 90 seconds till 1.7 hours post-LOCA per the requirements of NRC question 20 (below). The results are as follows:

- The worst-case containment pathway EAB dose was during the 0.4-2.4 hour period and resulted in a decrease in the EAB dose from 1.8988 Rem TEDE to 1.6975 Rem TEDE.
- The worst-case penetration room pathway EAB dose was during the 0.8-2.8 hour period and resulted in a decrease in the EAB dose from 0.1838 Rem TEDE to 0.1554 Rem TEDE.

These results will be incorporated into the MHA design basis as described in the response to NRC question 20 (below).

2. *It appears that the licensee assumes an instantaneous release of core activity for the "early in-vessel" phase, as opposed to the 1.3 hour, linear, release specified in the guidance of RG 1.183. Due to the credit being taken for time-dependent containment removal mechanisms, it is possible that the instantaneous methodology implemented may yield comparatively non-conservative results. Additionally, this instantaneous early in-vessel phase release methodology will likely skew the determination of the worst-case 2-hour EAB dose. Therefore, provide verification that the instantaneous early in-vessel phase release methodology that has been used is conservative when compared to the regulatory guidance.*

Response: We believe that regulatory guidance allows both ramp and step increases to model isotopic releases as described in Reference 2, Footnote 12. We initially used a step increase in the MHA calculation for ease of modeling the DF for spray removal of elemental iodine. A DF of 14.04 was calculated for spray removal of elemental iodine. Based on a spray removal rate of 14.816/hr, an effective removal time of 0.178 hours can be calculated for the step increase, starting at 0.5 hours post-LOCA. For a linear ramp increase, this methodology is inappropriate. A DF of 14.04 means that ~92.88% of elemental iodine must be removed before the sprays become ineffective on elemental iodine. For a linear ramp increase, 92.88% of the iodine is not injected into Containment until ~1.7 hours post-LOCA.

However, to respond to the above question, a sensitivity study was done and the MHA containment and vent stack pathway cases were rerun with a linear ramp isotopic release rate and with containment sprays effective on elemental iodine from 90 seconds till 1.7 hours post-LOCA. The original and updated results are as follows:

MHA Results Comparisons			
Results	EAB Rem	LPZ Rem	CR Rem
Containment Pathway			
Step	1.8988	0.4958	3.9682
Ramp	1.6975	0.4227	3.7957
Penetration Room Pathway			
Step	0.1838	0.0485	0.3968
Ramp	0.1554	0.0350	0.3699

Note that the isotopic ramp increase is less conservative than the original isotopic step increase. The 2 hour EAB doses for the ramp cases are worst-case taken over 0-30 days per the Regulatory Guidance provided in NRC question 19 above.

Since this NRC question notes that the regulatory guidance prefers a linear ramp increase, the MHA results are modified to incorporate the preferred NRC regulatory guidance. The new MHA results are as follows:

MHA Results			
Results	EAB Rem	LPZ Rem	CR Rem
Containment Pathway	1.6975	0.4227	3.7957
Penetration Room Pathway	0.1554	0.0350	0.3699
Refueling Water Tank Pathway	1.9676E-05	1.8560E-03	3.2979E-01
Hydrogen Purge Line Pathway	6.4918E-05	1.5283E-05	7.7048E-05

Containment Shine			0.0547
Plume Shine			0.0030
Control Room Filter Shine			0.0139
Total	1.8529	0.4595	4.5670
Regulatory Limits	25	25	5

Section 4.1.1 of Attachment 1 (Reference 1) is revised as follows:

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4.1.1 Maximum Hypothetical Accident (MHA)

Section 14.24 of the CCNPP UFSAR describes the MHA. The MHA is a non-mechanistic scenario which evaluates the Containments' capability to contain released radioisotopes. Safety system effectiveness is not considered; the quantity of radioisotopes released to the containment atmosphere is dependent on the power level (MWt) of the reactor. The criteria for this release are established so that the magnitude of the release bounds all credible accident releases.

Enclosure (1) contains the detailed MHA radiological consequences design basis calculation using the AST. The following supporting calculations are also provided: Atmospheric Dispersion Coefficient (X/Q) calculation (Enclosure 9), Source Terms calculation (Enclosure 10), and Primary and Secondary Isotopic calculations (Enclosure 12).

The MHA re-analysis for AST implementation considered the following radiological release pathways for offsite and Control Room doses.

- Containment pathway
- Hydrogen Purge line pathway
- Ventilation stack pathway
- Refueling Water Tank pathway
- Containment shine
- Plume shine
- Control Room Filter shine

Major assumptions and required plant modifications considered in the MHA re-analysis to meet regulatory requirements are:

- A bounding Control Room in-leakage value of 3,500 cfm,
- Modification of the CREVS to a nominal 10,000 cfm flow with 90 percent filtration efficiency for elemental and organic iodine and 99 percent for particulate iodine was credited,
- Installation of automatic isolation dampers and radiation monitors at Access Control HVAC Unit RTU-1 and Access Control Air Conditioning Unit 13 on the Auxiliary Building roof,
- Revision to the TS 3.4.15 limit for RCS activity from 1.0 μCi/gm to 0.5 μCi/gm,
- Revision to the TS 5.5.16 maximum allowable containment leakage rate, L_a, from 0.20 percent of containment air weight per day at P_a to 0.16 percent of containment air weight per day at P_a,