September 20, 2006

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3, ISSUANCE OF AMENDMENT RE: RECIRCULATION SPRAY SYSTEM (TAC NO. MC8327)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment No. 233 to Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 13, 2005, as supplemented by letters dated June 13 and August 14, 2006. The amendment revises the TS surveillance requirements for the recirculation spray system.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Victor Nerses, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures:
1. Amendment No. 233 to NPF-49
2. Safety Evaluation

cc w/encls: See next page

September 20, 2006

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Package Accession No.: **ML062220155**
Amendment Accession No.: **ML062220160**
TS(s) Accession No.: **ML062640464** * SE input provided - no major changes made

OFFICE	LPL1-2/PM	LPL1-2/PM	LPL1-2/LA	DE/EEEB/BC	DSS/SSIB/BC
NAME	REnnis:rsa	VNerses	CRaynor	GWilson*	MScott*
DATE	8/15/06	8/10/06	8/15/06	7/3/06	6/28/06
OFFICE	DIRS/ITSB/BC	DRA/AADB/BC	OGC/NLO	LPL1-2/BC (A)	
NAME	TKobetz	MKotzalas*	DRoth	BPoole	
DATE	8/22/06	8/8/06	9/6/06	9/14/06	

OFFICIAL RECORD COPY

Millstone Power Station, Unit No. 3

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DOMINION NUCLEAR CONNECTICUT, INC

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 233
License No. NPF-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Dominion Nuclear Connecticut, Inc. (the licensee) dated September 13, 2005, as supplemented by letters dated June 13 and August 14, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-49 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 233, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. Dominion Nuclear Connecticut, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to entering Mode 1 following refueling outage 3R11.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Brooke D. Poole, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 20, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 233

FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace the following page of the Appendix A, Technical Specifications with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3/4 6-13	3/4 6-13

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 233 TO FACILITY OPERATING

LICENSE NO. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated September 13, 2005, as supplemented by letters dated June 13 and August 14, 2006, Dominion Nuclear Connecticut, Inc. (DNC or the licensee) submitted an application requesting a license amendment for Millstone Power Station, Unit No. 3 (MPS3). The proposed amendment would revise the MPS3 Technical Specification (TS) Surveillance Requirements (SRs) for the recirculation spray system (RSS).

DNC determined that the proposed RSS TS changes, and an associated plant modification, were needed as part of the MPS3 resolution of Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance." The plant modification would alter the RSS pump circuitry by initiating the pump start from a reactor water storage tank (RWST) low-low level signal instead of pump start from a timer. The associated SR, TS 4.6.2.2.c, currently requires that each RSS be demonstrated to be operable at least once per 24 months by verifying that on a containment depressurization actuation (CDA) test signal, each RSS pump starts automatically after a 660 ±20 second time delay. The amendment would revise TS 4.6.2.2.c such that each RSS be demonstrated to be operable at least once per 24 months by verifying that on a CDA test signal, each RSS pump starts automatically after receipt of a RWST low-low level signal.

The supplements dated June 13 and August 14, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 25, 2005 (70 FR 61657).

2.0 REGULATORY EVALUATION

On September 13, 2004, the NRC issued Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors." The GL identified a potential susceptibility of recirculation flow paths and sump screens to debris blockage. The GL requested that addressees perform an evaluation of the emergency core cooling system (ECCS) and containment spray system (CSS) recirculation

functions in light of the information provided in the letter and, if appropriate, take additional actions to ensure system function. GL 2004-02 is part of the regulatory framework the NRC staff is using to address issues associated with GSI-191. The objective of GSI-191 is to ensure that post-accident debris blockage will not impede or prevent the operation of the ECCS and CSS in recirculation mode at pressurized water reactors during loss-of-coolant accidents (LOCAs) or other high-energy line break accidents for which sump recirculation is required.

The regulatory requirements which the NRC staff applied in its review included:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power plants,” which requires that the ECCS have the capability to provide long-term cooling of the reactor core following a LOCA.

- 10 CFR 50.67, “Accident source term” as supplemented by Regulatory Position C.4.4 of Regulatory Guide (RG), 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” and Standard Review Plan (SRP) 15.0.1, “Radiological Consequences Analyses Using Alternative Source Terms,” as they relate to radiological dose acceptance criteria.

- 10 CFR Part 50, Appendix A, Generic Design Criterion (GDC) 17, “Electric power systems,” which requires that nuclear power plants have an onsite and offsite electric power system to permit the functioning of structures, systems and components important to safety. The onsite system is required to have sufficient independence, redundancy and testability to perform its safety function, assuming a single failure, and the offsite system is required to be supplied by two independent circuits. In addition, this criterion requires provisions to minimize the probability of losing electric power from the remaining electric power supplies as the result of loss of power from the unit, the offsite transmission network, or the onsite power supplies.

- 10 CFR Part 50, Appendix A, GDC 18, “Inspection and testing of electric power systems,” which requires that electric power systems that are important to safety must be designed to permit appropriate periodic inspection and testing.

- 10 CFR Part 50, Appendix A, GDC 19, “Control room,” as it relates to maintaining the control room, in a safe condition under accident conditions by providing adequate protection against radiation.

- 10 CFR Part 50, Appendix A, GDC 38, “Containment heat removal,” which requires, among other things, that the containment pressure and temperature be maintained at acceptably low levels following a LOCA.

3.0 TECHNICAL EVALUATION

3.1 Background

As part of the MPS3 resolution of GSI-191, DNC identified that a plant modification, and associated TS change, was necessary for the RSS to increase the available margin to suction pipe flashing and for pump net positive suction head in order to accommodate the increase in

head loss due to potential debris blockage. The plant modification would alter the RSS pump circuitry by initiating the pump start from a RWST low-low level signal instead of pump start from a timer. The associated SR, TS 4.6.2.2.c, currently requires that each RSS be demonstrated to be operable at least once per 24 months by verifying that on a CDA test signal, each RSS pump starts automatically after a 660 ±20 second time delay. The amendment would revise TS 4.6.2.2.c such that each RSS be demonstrated to be operable at least once per 24 months by verifying that on a CDA test signal, each RSS pump starts automatically after receipt of a RWST low-low level signal.

The MPS3 CSS design includes two sets of pumps that reduce containment temperature and pressure. The sets of pumps included in the CSS are the quench spray system (QSS) pumps and the RSS pumps. After a LOCA, reactor coolant system (RCS) pressure will drop, resulting in a safety injection signal (SIS). Containment pressure will rise, resulting in a CDA signal. Upon receipt of the SIS, the charging pumps start injecting water into the RCS from the RWST. Upon receipt of the CDA signal, the QSS pumps also start drawing water from the RWST and spray that water into containment via spray headers. For the current MPS3 design, after a time delay of approximately 660 seconds, the RSS pumps will start drawing water from the containment emergency sump and will spray that water into containment to assist in lowering containment temperature and pressure.

For the current design, the licensee determined there would be a limited quantity of water on the containment floor (in the containment sump) for LOCAs. For the limiting case, there is approximately 1.5 inches of margin to suction line flashing. The licensee stated that this amount of margin is adequate to support the current licensing basis for operability of the MPS3 ECCS. The proposed modification (i.e., initiating RSS pump start from RWST low-low level) would increase the RSS pumps start delay time following a CDA signal in order to provide additional water level to the containment floor. The licensee stated that this change would ensure operability of the RSS pumps under the revised licensing basis to be established to resolve GSI-191.

3.2 Containment Pressure and Temperature Impact

The licensee performed an analysis of containment pressure and temperature with the proposed delay in RSS pump start time from the present timer setting to the estimated time of the RWST low-low level signal. The analysis was used to determine the containment pressure and temperature response for a large-break LOCA at various break locations and for various break sizes and single failure assumptions. The analysis showed that the peak pressure and temperature occurred 18 and 17.9 seconds after the accident, respectively. The proposed change does not affect either the time or the magnitude of either the peak pressure or peak temperature because both the existing and proposed RSS pump start times are significantly later than the time of containment peak pressure and temperature. During the RSS start delay, the QSS pumps continue to run and the analysis showed that containment pressure and temperature during this time do not exceed their peak values occurring at about 18 seconds after the accident.

The NRC staff reviewed the licensee’s analysis of the containment pressure and temperature to ensure that the licensee complies with the acceptance criteria for the ECCS and that the proposed changes do not have an adverse impact on the containment. The NRC staff concludes that the proposed changes are acceptable with respect to the requirements in

10 CFR 50.46 and GDC 38 based on review of the containment pressure and temperature analysis. Specifically, peak containment pressures and temperatures are not affected by the proposed changes. As such, the NRC staff concludes that the delay of the RSS pump start to the RWST low-low level signal will have no adverse effect on containment heat removal.

3.3 Emergency Diesel Generator (EDG) Starting and Loading Impact

The licensee stated that, at present, the residual heat removal system (RHS) (low head safety injection) pumps start on a SIS and are automatically secured when the RWST reaches its low-low level. Also, with the current design, the RSS pumps start approximately 11 minutes following a CDA and run for the duration of the accident. Thus, both RHS and RSS pumps run at the same time for part of the accident sequence. With the proposed design, the RSS pump start would be delayed until the RWST low-low level signal is received which is the same signal that stops the RHS pumps.

With the proposed modification, the RHS and RSS pumps will no longer operate at the same time. The RSS pumps in the current design have a staggered start on the timer following a CDA signal. In the proposed design, a time delay is provided between the start of two RSS pumps on each train to avoid overloading the EDG with high starting loads.

On June 13, 2006, the licensee provided a response to the NRC staff's request for additional information regarding the affected calculations. The NRC staff has reviewed the affected calculations and agrees with the licensee that the requested change will have no adverse impact on the EDG capability and, therefore, the proposed design change (RSS pumps start after RHS pump stops) is acceptable. The NRC staff also concludes that the proposed change will not affect the compliance of MPS3 with the requirements of GDC-17 and GDC-18.

3.4 Radiological Consequences Impact

In MPS3 Amendment No. 232 dated September 15, 2006, the NRC staff approved implementation of a full-scope application of an alternative source term (AST) methodology in accordance with 10 CFR 50.67. This safety evaluation (SE) addresses the impact of the proposed RSS TS change on previously-analyzed design-basis accident (DBA) radiological consequences, including the AST-based analyses, and the acceptability of the revised analysis results. The NRC staff evaluated the radiological consequences of the proposed amendment against the dose criteria specified in 10 CFR 50.67(b)(2); these criteria are 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 25 rem TEDE at the outer boundary of the low population zone (LPZ) for the duration of the postulated fission product release, and 5 rem TEDE in the control room (CR) for the duration of the postulated fission product release.

3.4.1 LOCA Analysis

The postulated DBA LOCA is the only MPS3 DBA affected by the proposed TS change, since only the LOCA analysis takes credit for operation of the RSS. The licensee revised the LOCA dose analysis, to reflect changes in timing for initiation of the RSS, in support of the proposed license amendment.

The radiological consequence DBA LOCA analysis is a deterministic evaluation based on the assumption of a major rupture of the primary RCS piping. The accident scenario assumes the deterministic failure of the ECCS to provide adequate core cooling, which results in a significant amount of core damage, as specified in RG 1.183. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design-basis accident analyses.

In the evaluation of the LOCA design-basis radiological analysis, the licensee included dose contributions from the following sources:

- Containment leakage plume
- ECCS component leakage
- RWST vent releases
- Shine from containment and the plume
- Shine from the CR filter loading

During a design-basis LOCA, it is assumed that the initial fission product release to the containment will last 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission products in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. The licensee used the LOCA source term release fractions, timing characteristics, and radionuclide grouping, as specified in RG 1.183. With the exception of the containment spray assumptions, as discussed in SE Section 3.4.1.1, all other assumptions are the same as previously approved.

3.4.1.1 Containment Sprays

The MPS3 design-basis LOCA analysis credits the use of containment sprays to remove elemental and particulate iodine from the containment atmosphere. Credit for the use of containment sprays for elemental and particulate iodine removal by the QSS was approved in Amendment No. 211, dated September 16, 2002 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML022470399), and November 25, 2002 (ADAMS Accession No. ML023290568). In Amendment No. 211, the NRC staff approved a QSS coverage value of 50.27 percent and a QSS effective initiation time of 70.2 seconds. The proposed revised analysis assumes that the percentage of containment that is covered by quench spray is 49.63 percent and that the QSS becomes effective at 71 seconds post-LOCA. In addition, the revised analysis proposes to credit the RSS for containment iodine removal at 2710 seconds post-LOCA, thereby increasing the sprayed coverage to 64.5 percent during the time when both spray systems are operating. However, the partitioning of the fission products source between the sprayed/unsprayed regions (0.4963/0.5037) remains constant. The mixing

rate during spray operation is assumed to be two turnovers of the unsprayed volume per hour, which is consistent with the value accepted by the NRC staff in Amendment No. 211.

The containment spray volume coverage fractions for the QSS and the combined QSS plus RSS periods of operations, based on conservative evaluations of the total containment spray volume and the coverage fractions, were accepted by the NRC staff in the AST amendment (ADAMS Accession No. ML061990135).

The licensee applied calculated spray removal rates until the QSS is secured at 6620 seconds. The RSS is assumed to be activated at 2710 seconds and is assumed to be operating for the duration of the accident. The revised analysis credits removal of elemental iodine due to sprays until the decontamination factor (DF) reaches a value of 200. Credit for the particulate iodine removal due to sprays is taken for as long as the RSS remains operating. For the time period during which sprays are assumed to be operating, the licensee calculated an elemental iodine DF of 20. The licensee calculated that a particulate iodine DF of 50 would be attained at 7362 seconds, at which time the calculated particulate removal rate was reduced by a factor of 10, in accordance with SRP 6.5.2 and RG 1.183.

A summary of the current and the proposed licensing basis is given below.

Parameter	**Current basis**	**Proposed basis**
Quench Sprays	Effective from 72.5 seconds to 7480 seconds	Effective from 71 seconds to 6620 seconds
Recirculation Sprays	Effective from 840 seconds (14 min) to 7480 seconds	Effective from 2710 seconds to 30 days
Recirculation Only Spray Coverage	not credited	1,102,000 ft^3
Particulate Iodine Removal Coefficients Quench Spray only QSS and RSS sprays RSS spray only	 12.73 per hour 16.14 per hour not credited	 12.37 per hour 14.11 per hour 7.77 per hour for DF<50 0.78 per hour for DF>50
Time at which elemental iodine DF = 200	not reached - sprays secured at 7480 seconds	2.636 hours
Time at which particulate iodine DF = 50	1.9 hours	2.045 hours
Start time of ECCS leakage	640 seconds	2530 seconds

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose acceptance criteria specified in 10 CFR 50.67. The licensee's calculated dose results are shown in Table 1.

The NRC staff performed independent confirmatory dose calculations for the LOCA event using the NRC-sponsored radiological consequence computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.03, as described in NUREG/CR-6604. The RADTRAD code, developed by the Sandia National Laboratories for the NRC, estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff's independent evaluation of the radiological consequences resulting from the revised assumptions confirmed that the net result is a reduction in the predicted doses from the DBA LOCA. Therefore, the proposed revised assumptions are acceptable.

3.4.2 Radiological Consequences Summary and Conclusion

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impact of the proposed amendment on the radiological consequences of a DBA LOCA. The NRC staff performed independent confirmatory dose evaluations to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative guidance in RG 1.183. The assumptions found acceptable to the NRC staff are presented in Tables 2 and 3.

The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in 10 CFR 50.67 and SRP 15.0.1. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The NRC staff further finds reasonable assurance that the MPS3 TSs, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters.

Based on the above considerations, the NRC staff concludes that the proposed license amendment is acceptable with respect to the radiological consequences of a DBA.

3.5 Technical Evaluation Conclusion

Based on the considerations discussed in SE Sections 3.1 through 3.4, the NRC staff finds the proposed changes to TS 4.6.2.2.c to be acceptable.

The licensee stated that the TS Bases would also be revised to describe the acceptability of the 24-month frequency for checking the RSS pump start on a RWST low-low level signal. The NRC staff considered the proposed TS Bases changes as information only. The NRC staff did not review or make a finding with respect to these changes.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official agreed with the NRC staff's conclusion as stated in SE Section 6.0.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 61657). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: A. Pal
R. Reyes
A. Drozd
R. Ennis

Date: September 20, 2006

Table 1
MPS3 Radiological Consequences Expressed as TEDE (rem)

	EAB	LPZ	CR
MPS3 LOCA Analysis Results	7.5E+00	1.8E+00	1.9E+00
Dose Criteria	2.5E+01	2.5E+01	5.0E+00

Table 2
MPS3 Control Room Data and Assumptions

CR effective volume	2.38E+05 ft^3
Normal CR intake flow rate prior to isolation	1595 cfm
Unfiltered inleakage during periods of neutral pressure	350 cfm
Unfiltered inleakage during periods of positive pressure	100 cfm
Control room emergency ventilation system (CREVS) recirculation flow rate	666 cfm
CREVS pressurization flow rate	230 cfm
Response time for CR inlet radiation monitor to generate control building isolation (CBI) signal	5 seconds
Response time for CR to isolate upon receipt of CBI	5 seconds
Time allotted for delay of control room envelope pressurization system (CREPS)	1 minute
Time allotted for CREPS discharge to the CR (CREPS is not credited in any dose analyses)	60 minutes
Time allotted for operator action to align CREVS after CREPS discharge	40 minutes
Total time allotted to place CREVS in service (summation of the 3 preceding time intervals)	101 minutes after CBI signal
Filter efficiencies for CREVS	90% elemental 90% aerosol 70% organic
Containment wall thickness	4.5 ft concrete
Containment dome thickness	2.5 ft concrete
Control building wall thickness	2 ft concrete
Control room ceiling thickness	8 inches concrete
Control building roof thickness	1 ft-10 in concrete
CR occupancy factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4
Breathing rate for CR dose analyses	3.5E-04 m^3/sec

Table 3
MPS3 Data and Assumptions for the LOCA

Containment free air volume	2.35E+06 ft^3
Containment leak rate	0.3% weight% per day (L_a)
Containment bypass leak rate	0.06L_a
Containment leak rate reduction	50% after 24 hours (offsite analyses)
	50% after 1 hour (CR analysis)
Secondary containment drawdown time	2 minutes
Iodine chemical form in containment atmosphere	95% cesium iodide
	4.85% elemental iodine
	0.15% organic iodine
Iodine chemical form in the sump and RWST	97% elemental
	3% organic
Containment sump pH	$ 7
Supplementary leak collection and release system filter efficiency	95% all iodines and particulates
Auxiliary building filter efficiency	95% all iodines and particulates
QSS effective operation period	71 - 6620 seconds
RSS start time	Low-low RWST signal (2710 seconds)
RSS effective time	30 days
Elemental iodine removal coefficient	20 per hour (0.01972 - 2.636 hrs)
QSS particulate iodine removal coefficient	DF < 50: 12.37 (0.01972 - 0.7528 hr)
Particulate iodine removal coefficient for QSS and RSS	DF < 50: 14.11 (0.7528 - 1.839 hrs)
RSS particulate iodine removal coefficient	DF < 50: 7.77 (1.839 - 2.636 hrs)
	DF > 50: 0.78 (2.636 - 720 hrs)
QSS containment coverage volume	1,166,200 ft^3
QSS and RSS containment coverage volume	1,515,858 ft^3
RSS containment coverage volume	1,102,000 ft^3
ECCS leakage outside containment	4730 cc/hr
Minimum available RWST volume	1,072,886 gallons
Minimum QSS auto trip value	47,652 gallons
RWST maximum fill volume	1,206,644 gallons