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Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

Operated by Nuclear Management Company LLC

October 19, 2000
NG-00-1589

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Technical Specification Change Request (TSCR-037): "Alternative Source Term"

Reference NG-00-1504, "Supplement to Duane Arnold Energy Center Environmental Report", dated September 22, 2000

File: A-117

Dear Sir(s):

In accordance with the Code of Federal Regulations, Title 10, Section 50.67 and 50.90, the Nuclear Management Company (NMC) requests review and approval of a full scope application of an alternative source term for the Duane Arnold Energy Center (DAEC). DAEC also requests an amendment to the DAEC Technical Specifications which implements various assumptions in the Alternative Source Term analyses. The amendment request also seeks changes to secondary containment operability requirements which are justified by the analyses.

The Alternative Source Term analyses follow the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" with the following exception:

- The analyses were performed using the values provided in the draft Regulatory Guide for radionuclide fission product group fractions assumed to be in the fuel-cladding gap during Control Rod Drop Accident (CRDA) events. This was a conservative assumption for all groups except the alkali metals where the value increased from

A001

0.10 to 0.12. Overall, the use of the draft Regulatory Guide assumptions results in calculated doses that are conservative by as much as one third compared to the values in the final Regulatory Guide.

Recognizing the conservative nature of the deviations taken from the final Regulatory Guide 1.183, DAEC does not intend to conduct conforming analyses at this time. However, at such time these analyses are reperformed to support future projects, these analyses will be brought in compliance.

The analyses assume an uprated power level of 1,912 MWth (compared to the current licensed limit of 1,658 MWth) and the use of General Electric GE-14 fuel which utilizes higher enrichments and allows higher burnups than those fuel types for which DAEC is currently licensed. No actual increase in licensed power levels or change in fuel types is being sought by this submittal. That request is being docketted separately. Other changes/additions to the DAEC licensing basis that result from the analyses include:

- A "Positive Pressure Period" during drawdown of secondary containment negative pressure is now assumed to occur at the beginning of a loss of coolant (LOCA) event.
- Credit is now taken for the automatic isolation of the Control Room ventilation on high radiation levels during a LOCA where none was taken previously. The enclosed Technical Specification amendment request seeks to reduce the allowed value for the isolation from ≤ 50 mR/hr to ≤ 5 mR/hr.
- Credit for use of Standby Liquid Control (SBLC) to buffer suppression pool pH is now credited to prevent iodine re-evolution during a postulated radiological release as defined in NUREG-1465.
- Emergency Core Cooling System leakage is now assumed to occur during accident conditions.
- New offsite atmospheric dispersion factors (χ/Q's) were calculated.
- The Technical Specification amendment request seeks to lower the dose equivalent I-131 specific activity limits currently provided from 1.2 μCi/ml to 0.2 μCi/gm and from 12.0 μCi/ml to 2.0 μCi/gm to conform with the analysis.

Additionally, because the Alternative Source Term no longer assumes secondary containment operability during fuel handling accidents (FHAs), the enclosed Technical Specification amendment request seeks to remove secondary containment operability requirements (including Secondary Containment Isolation Valves/Dampers, secondary containment instrumentation and Standby Gas Treatment) during core alterations and while moving irradiated fuel.

The DAEC Operations Committee and the Safety Committee have reviewed this application. A copy of this submittal, including the enclosed evaluation of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to 10 CFR 50.91. Affected Technical Specification Bases pages have been marked up and included to assist your review of the amendment application. DAEC requests review and approval of this request by April 30, 2001 to support the Spring refueling outage.

This letter is true and accurate to the best of my knowledge and belief.

NUCLEAR MANAGEMENT COMPANY, LLC

By Gary Van Middlesworth
Gary Van Middlesworth
DAEC Site General Manager

State of Iowa
(County) of Linn

Signed and sworn to before me on this 19th day of October, 2000,

by Gary Van Middlesworth.

Nancy S. Franck
Notary Public in and for the State of Iowa

NANCY S. FRANCK

9-28-01
Commission Expires

Attachments:
1) EVALUATION OF CHANGE PURSUANT TO 10 CFR SECTION 50.92
2) PROPOSED CHANGE TSCR-037 TO THE DUANE ARNOLD ENERGY CENTER TECHNICAL SPECIFICATIONS
3) ENVIRONMENTAL CONSIDERATION
4) SAFETY ASSESSMENT

cc: L. Sueper
M. Wadley
B. Mozafari (NRC-NRR)
J. Dyer (Region III)
D. McGhee (State of Iowa)
NRC Resident Office
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EVALUATION OF CHANGE PURSUANT TO 10 CFR SECTION 50.92

Background:

Duane Arnold Energy Center (DAEC) requests review and approval to revise its licensing basis to utilize an alternative accident source term as described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants". The current licensing basis utilizes a source term determined in accordance with TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", 1962. The analyses in support of this alternative source term were developed in accordance with the guidance provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and Standard Review Plan (SRP) Section, 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms," with exceptions. This submittal also requests review and approval of an amendment to the DAEC Technical Specifications which implements certain assumptions used in the Alternative Source Term analyses as well as changes to the secondary containment operability requirements justified by the analyses. The specific changes being sought are summarized below.

Nuclear Management Company, LLC, Docket No. 50-331,
Duane Arnold Energy Center, Linn County, Iowa
Date of Amendment Request: October 19, 2000

Description of Changes:

The definition of Dose Equivalent Iodine-131 in section 1.1 will be revised to reference Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993. The word "thyroid" is removed.

The setpoint for the Allowable Value of SR 3.3.7.1.3, channel calibration of Control Building Air Intake Radiation Monitor, will be reduced from ≤ 50 mR/hr to ≤ 5 mR/hr.

The dose equivalent I-131 specific activity limits stated in the LCO 3.4.6 action statements will be lowered from 1.2 μCi/ml to 0.2 μCi/gm and from 12.0 μCi/ml to 2.0 μCi/gm.

The requirement to maintain Secondary Containment, Secondary Containment Isolation Valves/Dampers, Secondary Containment Instrumentation, and Standby Gas Treatment operable during fuel movement of irradiated fuel assemblies and during core alterations will be removed.

Various references to 10 CFR 100, in both the Technical Specifications and Bases, will be changed to 10 CFR 50.67 to reflect adoption of the Alternative Source Term.

Basis for proposed No Significant Hazards Consideration:

The Commission has provided standards (10 CFR Section 50.92(c)) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

After reviewing this proposed change, we have concluded:

1) The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Alternative Source Term and those plant systems affected by implementing the setpoints and action levels specified in the analyses are not assumed to initiate design basis accidents. The Alternative Source Term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated the new source term is an input to evaluate the consequence. The implementation of the Alternative Source Term has been evaluated in revisions to the analyses of the limiting design bases accidents at DAEC. Based on the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the Alternative Source Term. This guidance is presented in NUREG 1465, 10 CFR 50.67, associated Regulatory Guide 1.183, and Standard Review Plan (SRP) Section, 15.0.1. Since secondary containment operability is not assumed for the fuel handling accident (FHA), the consequences of eliminating the requirements for secondary containment operability, secondary containment isolation valves/dampers, secondary containment instrumentation and the Standby Gas Treatment system during fuel movement or core alterations will not increase the effects of a FHA beyond those evaluated in the Alternative Source Term analysis. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

2) The proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The Alternative Source Term and those plant systems affected by implementing the setpoints and action levels specified in the analyses do not initiate design basis accidents.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3) The proposed amendment will not involve a significant reduction in a margin of safety.

The changes proposed are associated with the implementation of a new licensing basis for DAEC. Approval of the basis change from the original source term developed in accordance with TID-14844 to a new alternative source term as described in NUREG-1465 is requested by this submittal. The results of the accident analyses revised in support of this submittal, and the requested Technical Specification changes, are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies. Safety margins and analytical conservatisms have been evaluated and are satisfied. The analyzed events have been carefully selected and margin has been retained to ensure that the analyses adequately bound all postulated event scenarios. The dose consequences of these limiting events are within the acceptance criteria also found in the latest regulatory guidance. This guidance is presented in NUREG 1465, in the approved rulemaking for 10 CFR 50.67, and in the associated Regulatory Guide 1.183.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limit. Specifically, the margin of safety for these accidents is considered to be that provided by meeting the applicable regulatory limit, which, for most events, is conservatively set below the 10 CFR 50.67 limit. With respect to the control room personnel doses, the margin of safety (the difference between the 10 CFR 50.67 limits and the regulatory limit defined by 10 CFR50, Appendix A, Criterion 19 (GDC 19)) continues to be satisfied.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, they are considered to not result in a significant reduction in a margin of safety.

Based upon the above, we have determined that the proposed amendment will not involve a significant hazards consideration.

Local Public Document Room Location: Cedar Rapids Public Library, 500 First Street SE, Cedar Rapids, Iowa 52401

Attorney for Licensee: Al Gutterman; Morgan, Lewis & Bockius, 1800 M Street NW, Washington, D.C. 20036-5869

PROPOSED CHANGE TSCR-037 TO THE DUANE ARNOLD ENERGY CENTER TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend the Technical Specifications as summarized below.

SUMMARY OF CHANGES

Page	Description of Change
1.1-3	The definition of Dose Equivalent Iodine-131 replaces the reference to TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" with Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993. The word "thyroid" is removed.
3.3-65	Remove footnote (b) ("During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.") from Table 3.3.6.2-1.
3.3-71	Reduces the ALLOWABLE VALUE of SR 3.3.7.1.3, channel calibration of Control Building Air Intake Radiation Monitor, from ≤ 50 mR/hr to ≤ 5 mR/hr.
3.4-13,14	The dose equivalent I-131 specific activity limits stated in LCO 3.4.6, its CONDITIONS, and SURVEILLANCE REQUIREMENTS will be lowered from 1.2 μCi/ml to 0.2 μCi/gm and from 12.0 μCi/ml to 2.0 μCi/gm.
3.6-35	Remove "During movement of irradiated fuel assemblies in the secondary containment" and "During CORE ALTERATIONS," from the APPLICABILITY of LCO 3.6.4.1. Remove "during movement of irradiated fuel assemblies in the secondary containment during CORE ALTERATIONS, or" from Condition C. Delete REQUIRED ACTION C.1, "Suspend movement of irradiated fuel assemblies in the secondary containment. AND" and delete associated COMPLETION TIME.

3.6-36 Remove REQUIRED ACTION C.2, "Suspend CORE ALTERATIONS. AND" and associated COMPLETION TIME. Renumber remaining REQUIRED ACTION from "C.3" to "C.1".

3.6-37 Remove "During movement of irradiated fuel assemblies in the secondary containment" and "During CORE ALTERATIONS," from the APPLICABILITY of LCO 3.6.4.2.

3.6-39 Remove "movement of irradiated fuel assemblies in the secondary containment during CORE ALTERATIONS, or during" from CONDITION D. Delete REQUIRED ACTION D.1 (excluding note) and D.2. Delete associated COMPLETION TIMEs. Renumber D.3 to D.1.

3.6-41 Remove "During movement of irradiated fuel assemblies in the secondary containment" and "During CORE ALTERATIONS," from the APPLICABILITY of LCO 3.6.4.3. Remove "movement of irradiated fuel assemblies in the secondary containment during CORE ALTERATIONS, or during" from Condition C.

3.6-42 Delete REQUIRED ACTION C.2.1 and C.2.2 and associated COMPLETION TIMEs. Renumber REQUIRED ACTION C.2.3 to C.2.1. Remove "during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or" from CONDITION E. Delete REQUIRED ACTION E.1 (excluding note) and associated COMPLETION TIME.

3.6-43 Delete REQUIRED ACTION E.2 and associated COMPLETION TIME. Renumber ACTION E.3 to E.1.

Note: The following Bases pages are included for information only.

B 3.4-33-36 Change references to 10 CFR 100 (or 10 CFR 100.11, 1973) to 10 CFR 50.67. Change references from 1.2 μCi/ml to 0.2 μCi/gm and from 12.0 μCi/ml to 2.0 μCi/gm. Replace "a value from a parametric evaluation of typical site locations. This limit is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site." with "plant specific analysis of the radiological consequences of a Main Steam Line Break DBA." Replace "thyroid and whole body" with "Total Effective Dose Equivalent (TEDE)."

B 3.6-90,91 Change references from 10 CFR 100 to 10 CFR 50.67.

B 3.7-18	Replace "whole body dose or its equivalent to any part of the body" with "total effective dose equivalent (TEDE)".
B 3.7-30, 32, 37, and 39	Change references from 10 CFR 100 (or 10 CFR 100.11, 1973) to 10 CFR 50.67. Change "NUREG 0800" to "Regulatory Guide 1.183".
B 3.9-19	Replace reference in BACKGROUND from 10 CFR 100 to Regulatory Guide 1.183.
B 3.9-21	Change REFERENCE 2 from 10 CFR 100.11 to Regulatory Guide 1.183.

ENVIRONMENTAL CONSIDERATION

10 CFR Section 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released off site; and (3) result in a significant increase in individual or cumulative occupational radiation exposure. DAEC has reviewed this request and determined that the Alternative Source Term and proposed Technical Specification amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9) for the following reasons:

1. As demonstrated in Attachment 1 to this letter, the Alternative Source Term and proposed amendment do not involve a significant hazards consideration.

2. Adoption of the Alternative Source Term and the Technical Specification changes which implement certain conservative assumptions in the Alternative Source Term analyses will not result in modifications to the plant or changes in its operation which could alter the type or amount of effluents that may be released offsite.

3. The Alternative Source Term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated the Alternative Source Term is an input to evaluate the consequence. The implementation of the Alternative Source Term has been evaluated in revisions to the analyses of the limiting design bases accidents at DAEC and with consideration of the supplement to the DAEC Environmental Report (Reference 1) which was docketted earlier. Based on the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these limiting events are within the regulatory guidance provided by the NRC for use with the Alternative Source Term. Thus, there will be no significant increase in either individual or cumulative occupational radiation exposure.

1.1 Definitions (continued)

CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/ml), that alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."~~ Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989 and FGR 12, "External Exposure to Radionuclides in Air, Water, and Soil," 1993.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME	The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint to actuation of the breaker secondary (auxiliary) contact. 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.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ 165.6 inches
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 2.2 psig
3. Reactor Building Exhaust Shaft - High Radiation	1,2,3, (a)~~, (b)~~	1	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 12.8 mR/hr
4. Refueling Floor Exhaust Duct - High Radiation	1,2,3, (a)~~, (b)~~	1	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 10.6 mR/hr

(a) During operations with a potential for draining the reactor vessel.

~~(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.~~

SURVEILLANCE REQUIREMENTS

----------NOTE----------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other channel is OPERABLE.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ ~~50~~ 5 mR/hr.	24 months
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ ~~1.2~~ 0.2 μCi/~~ml~~. g·m

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity > ~~1.2~~ 0.2 μCi/~~ml~~ g·m and ≤ ~~12.0~~ 2.0 μCi/~~ml~~ g·m DOSE EQUIVALENT I-131.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
OR	B.2.1 Isolate all main steam lines.	12 hours
Reactor Coolant specific activity > ~~12.0~~ 2.0 μCi/~~ml~~ g·m Dose EQUIVALENT I-131.	OR	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2.1 Be in MODE 3.	12 hours
	AND	
	B.2.2.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is ≤ ~~1.2~~ 0.2 μCi/~~ml~~ gm.	7 days

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During Operations with a Potential for Draining the Reactor Vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in the secondary containment. AND	~~Immediately~~ (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	~~C.2 Suspend CORE ALTERATIONS~~	~~Immediately~~
	~~AND~~	
	C.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed.	31 days
SR 3.6.4.1.2	----------NOTE---------- Doors in high radiation areas may be verified by administrative means. ---------- Verify that either the outer door(s) or the inner door(s) in each secondary containment access opening are closed.	31 days
SR 3.6.4.1.3	Verify each SBGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment at a flow rate ≤ 4000 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves/Dampers (SCIV/Ds)

LCO 3.6.4.2 Each SCIV/D shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During Operations with a Potential for Draining the Reactor Vessel (OPDRVs).

ACTIONS

----------NOTES----------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIV/Ds.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV/D inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve/damper, closed manual valve, or blind flange. AND	8 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	D.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in the secondary containment.	~~Immediately~~
	AND D.2 Suspend CORE ALTERATIONS.	~~Immediately~~
	AND D.3 Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SBGT) System

LCO 3.6.4.3 Two SBGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment~~,
~~During CORE ALTERATIONS~~,
During Operations with a Potential for Draining the Reactor Vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SBGT subsystem inoperable.	A.1 Restore SBGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during ~~movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.~~	----------NOTE---------- LCO 3.0.3 is not applicable. ---------- C.1 Place OPERABLE SBGT subsystem in operation. OR	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend movement of irradiated fuel assemblies in secondary containment. AND C.2.2 Suspend CORE ALTERATIONS. AND C.2.3 Initiate action to suspend OPDRVs.	~~Immediately~~ ~~Immediately~~ Immediately
D. Two SBGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately
E. Two SBGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Suspend movement of irradiated fuel assemblies in secondary containment. AND	~~Immediately~~ (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	~~E.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	~~E.3~~ Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SBGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	----------NOTE---------- When a SBGT subsystem is placed in an inoperable status solely for the performance of VFTP testing required by this Surveillance on the other subsystem, entry into associated Conditions and Required Actions may be delayed for up to 1 hour. ---------- Perform required SBGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation in normal operating conditions, the reactor coolant acquires radioactive materials due to release of fission products from "tramp" uranium on the outside of the fuel cladding and activation of corrosion products in the reactor coolant. There is the potential for fuel leaks into the reactor coolant that will contribute to the level of radioactive materials due to release of fission products. These radioactive materials in the reactor coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure that in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR ~~100~~ (Ref. 1). 50.67

This LCO contains iodine specific activity limits. The iodine isotopic activities per ~~milliliter~~ gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR ~~100~~ limit. 50.67

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the UFSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a Main Steam Line Break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

primary coolant ensure that the 2 hour ~~thyroid and whole body~~ doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR ~~100.~~ 50.67

Total Effective Dose Equivalent (TEDE)

The limit on specific activity is a value from a parametric evaluation of typical site locations. This limit is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

plant specific analyses of the radiological consequences of a Main Steamline Break DBA.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to ≤ ~~1.2~~ 0.2 μCi/~~ml~~ gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR ~~100~~ 50.67 limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ ~~12.0~~ 2.0 μCi/~~ml~~ gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ ~~1.2~~ 2.0 ~~μCi/ml~~ gm within 48 hours, or if at any time it is > ~~12.0~~ 0.2 ~~μCi/ml~~ gm, it must be determined at least once every 4 hours and all the main steam lines, including the main steam line drains, must be isolated within 12 hours. Isolating the main steam lines including drains, precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR ~~100~~ 50.67 during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without

(continued)

BASES

ACTIONS

B.1, B.2.1, B.2.2.1, and B.2.2.2 (continued)

challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The analysis is performed using filtrate from a 0.45μ filter. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. ~~10 CFR 100.11, 1973.~~ 10 CFR 50.67
2. UFSAR, Sections 15.6.5 and 15.10.3.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SBGT) System

BASES

BACKGROUND

The SBGT System is required by the UFSAR (Ref. 1). The function of the SBGT System is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The SBGT System consists of two fully redundant subsystems, each with its own set of ductwork, dampers, charcoal filter train, and controls.

Each charcoal filter train consists of (components listed in order of the direction of the air flow):

a. A demister or moisture separator;

b. An electric heater;

c. A prefilter;

d. A High Efficiency Particulate Air (HEPA) filter;

e. A charcoal adsorber;

f. A second HEPA filter; and

g. A centrifugal fan.

HEPA filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place testing of the HEPA filters and charcoal adsorbers is performed under the DAEC Ventilation Filter Testing Program (ITS 5.5.7). If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR ~~100~~ 50.67 guidelines for the accidents analyzed, as the UFSAR Section 15.6.6 for the loss-of-coolant accident shows compliance with

(continued)

BASES

BACKGROUND (continued)

10 CFR ~~100~~ 50.67 guidelines with an assumed efficiency of 99% for the adsorber. Operation of the fans significantly different from the design flow envelope will change the removal efficiency of the HEPA filters and charcoal adsorbers.

The sizing of the SBGT System equipment and components is based on the results of an infiltration analysis, as well as an exfiltration analysis of the secondary containment. The internal pressure of the SBGT System boundary region is maintained at a negative pressure of at least 0.25 inches water gauge (as determined by averaging pressure readings from different faces of the Secondary Containment boundary) when the system is in operation, which represents an internal pressure that ensures zero exfiltration of air from the building when exposed to calm wind conditions (< 15 mph). Maintaining a negative pressure of 0.25 inches water gauge under calm wind conditions ensures a negative pressure under worst case conditions. Therefore, a negative pressure of 0.25 inches water gauge includes some margin to a negative pressure that ensures zero exfiltration.

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SBGT System automatically starts and operates in response to secondary containment isolation actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both charcoal filter train fans start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

APPLICABLE SAFETY ANALYSES

The design basis for the SBGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents (Ref. 3). For all events analyzed, the SBGT subsystem is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.4 Standby Filter Unit (SFU) System

BASES

BACKGROUND

The SFU System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of the SFU System includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of outside supply air. Each subsystem consists of a demister, an electric heater, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a fan, and the associated ductwork and dampers. Demisters remove water droplets from the airstream. HEPA filters remove particulate matter, which may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

The SFU System is a standby system, parts of which also operate during normal unit operations to maintain the control room environment. Upon receipt of the initiation signal (indicative of conditions that could result in radiation exposure to control room personnel), the SFU System automatically starts and a system of dampers isolates the control building to prevent infiltration of contaminated air into the control room. Outside air is taken in at the normal ventilation intake and is passed through one of the charcoal adsorber filter subsystems for removal of airborne radioactive particles before being mixed with the recirculated air. The air (outside and/or recirculated) is cooled by Air Conditioning (AC) units supplied by the Control Building Chillers (CBCs). The SFUs and AC units share common ductwork such that either SFU may supply outside air to either AC unit. However, the CBCs and AC units are addressed as part of LCO 3.7.5, "Control Building Chiller System."

The SFU System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA without exceeding 5 rem ~~whole body dose or its equivalent to any part of the body~~ total effective dose equivalent (TEDE). A single SFU subsystem will

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensible gases are collected in the condenser, then exhausted through the Steam Jet Air Ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the 30 minute holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Steam-Line Break Accident (Roof Top Release). This analysis assumes the concentrations of radionuclides in the reactor water are those associated with an assumed stack gas release limit of 1,000,000 μCi/sec (1.0 Ci/sec)(Ref. 3).

The main condenser offgas limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Restricting the gross radioactivity rate of noble gases from the main condenser (Ref. 1) provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part ~~100~~ 50.67 (Ref. 2) in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50. The offgas

(continued)

BASES

ACTIONS

B.1, B.2, B.3.1, and B.3.2 (continued)

power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by ≥ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable, based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

1. UFSAR, Section 11.3.3.
2. 10 CFR ~~100.~~ 50.67
3. UFSAR, Section 15.10.3.

B 3.7 PLANT SYSTEMS

B 3.7.8 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the UFSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 15.7.1 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are well below the guideline limits of 10 CFR ~~100~~ 50.67 (Ref. 3) and meet the exposure guidelines of ~~NUREG-0800~~ Regulatory Guide 1.183 (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in UFSAR, Section 15.7.1.4 (Ref. 5).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the UFSAR, Section 9.1.2.3.3.4 (Ref. 6). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 9.1.2.
2. UFSAR, Section 15.7.1.
3. 10 CFR ~~100.~~ 50.67
4. ~~NUREG-0800, Section 15.7.4, Revision 1, July 1981.~~
5. UFSAR, Section 15.7.1.4. Regulatory Guide 1.183
6. UFSAR, Section 9.1.2.3.3.4.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Ref. 1). Sufficient iodine activity would be retained to limit offsite doses from the accident to less than ~~10 CFR 100~~ limits. (Regulatory Guide 1.183)

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment. A minimum water level of 23 ft allows a decontamination factor of 100 to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory.

Analysis of the fuel handling accident inside containment is described in Reference 1. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained within allowable limits (Ref. 2).

RPV water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV, ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. This verification can be performed using the normal RPV water level indication, which is referenced to Top of Active Fuel (TAF). Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 1).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. UFSAR, Section 15.7.1.
2. ~~10 CFR 100.11.~~ Regulatory Guide 1.183

Safety Assessment

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ACRONYMS AND ABBREVIATIONS

ACP	Administrative Control Procedure
AL	Analytical Limit
AST	Alternate Source Term
BOP	Balance of Plant
BWROG	Boiling Water Reactors Owners' Group
CR	Control Room
CRDA	Control Rod Drop Accident
DAEC	Duane Arnold Energy Center
DBA	Design Basis Accident
DEB	Double Ended Break
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EQ	Equipment Qualification
ESF	Engineered Safety Features
FHA	Fuel Handling Accident
HEPA	High Efficiency Particulate Air
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
MELLLA	Maximum Extended Load Line Limit Analysis
MSLBA	Main Steam Line Break Accident
MSLRM	Main Steam Line Radiation Monitor
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OL	Operating License
PASS	Post-Accident Sampling System
PPP	Positive Pressure Period
PUSAR	Power Uprate Safety Analysis Report
RPV	Reactor Pressure Vessel
SER	Safety Evaluation Report
SBGT	Standby Gas Treatment System
SFU	Standby Filter Unit
SJAE	Steam Jet Air Ejector
SLCS	Standby Liquid Control System
T-G	Turbine - Generator
TS	Technical Specifications

TSC	Technical Support Center

1. INTRODUCTION

1.1 Evaluation Overview and Objective

The objective of this evaluation is to document DAEC implementation of the Alternative Source Terms (AST) described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." (Reference 3) The implementation of AST is governed by 10 CFR 50.67 (Reference 2) and the guidelines of the Standard Review Plan (SRP) section 15.0-1 "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 4) and Regulatory Guide (RG) 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." (Reference 39). Portions of the analyses were conducted using the guidelines of the Draft Guide DG-1081 (Reference 5) finalized in Reference 39.

DAEC has elected to perform a full scope implementation of the AST as defined in section 1.2.1 of Reference 4. The implementation consists of the following steps:

1. identification of the alternative source term based on plant specific analysis of the fission product inventory,

2. calculation of the release fractions for the four BWR Design Basis Accidents (DBA),

3. analysis of the atmospheric dispersion for various radiological propagation pathways,

4. calculation of deposition and removal mechanisms,

5. calculation of offsite, control room, and emergency response facility personnel doses Total Effective Dose Equivalent doses, and

6. evaluation of other related design and licensing bases, such as NUREG-0737 and Equipment Qualification.

In order to support the DAEC application for an extended power uprate, the analyses have been performed assuming reactor operation at a thermal power of 1950 MWt (102% of 1912 MWt). This results in a conservative overestimation of fission product releases for operation at current licensed power of 1658 MWt.

1.2 Changes to DAEC Design and Licensing Bases

The following changes to the DAEC design and licensing bases were considered:

1. A bounding plant-specific fission product inventory was developed.
2. The AST methodology was adopted for the composition and timing of radiation releases.
3. Consideration of a positive pressure period during drawdown of secondary containment negative pressure was considered.
4. Atmospheric dispersion factors were reanalyzed for existing pathways and for new pathways.
5. ECCS Leakage was considered.
6. Operation of the Standby Gas Treatment System (SGTS) was not credited for the Fuel Handling Accident (FHA),
7. Changes were made to the assumed value for in-leakage to the CR and TSC,
8. Credit is taken for delayed automatic isolation of the control room on ventilation inlet high radiation.

1.3 Deviations from Regulatory Guidelines

Work had been completed on some analyses using the guidelines of DG-1081. Changes in the guidelines were included in the final version of RG 1.183. As a result, the following deviation was taken from the final RG at this time:

- Non-LOCA Fission Product release fractions for the CRDA were used from DG-1081 rather than from RG 1.183.Fractions. The release fractions for all isotope groups are more conservative in the DG except for the fraction for alkali metals which is 0.10 compared to a release fraction of 0.12 in RG 1.183. The overall effect of this deviation is a conservative over prediction of personnel doses by about one-third.

1.4 Summary

The radiological consequences of the plant's DBA accidents were analyzed for impact from and AST implementation assuming plant operation at 1,912 MWth. Implementation of the AST as the plant radiological consequence analyses licensing basis

requires a license amendment per the requirements of 10CFR50.67. The analyses demonstrate the offsite, Control Room (CR), and Technical Support Center (TSC) post accident doses remain under regulatory limits. An evaluation of compliance to NUREG-0737 demonstrates that the plant continues to meet acceptance criteria for post-accident conditions. An evaluation of the impact of post-accident equipment doses from operation at 1912 MWt and using the interim guidance of SECY-99-240 (Reference 5) and Reference 4 concluded that the existing analysis remains bounding. The interim guidance was provided pending issuance of a generic safety issue dealing with dose contributions from fission product cesium in sump water.

2. EVALUATION

2.1 Scope

2.1.1 Accident Radiological Consequence Analyses

The following accident analyses documented in the DAEC UFSAR (Reference 41) were addressed using methods and input assumptions consistent with the AST:

- UFSAR Section 15.4.7, Control Rod Drop Accident (CRDA)
- UFSAR Section 15.6.5, Main Steam Line Break Accident (MSLBA)
- UFSAR Section 15.6.6, Loss of Coolant Accident (LOCA), including Control Room (CR) habitability analysis discussed in UFSAR Sections 6.4 and 6.7.
- UFSAR Section 15.7.1, Refueling Accident

The analyses were based on assumed operation at 1912 MWt. In addition, the analysis assumptions followed the guidelines prescribed in SRP 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 4), Draft Regulatory Guide DG-1081 "Alternative Radiological Source Terms for Evaluating the Radiological Consequences of Design Basis Accidents at Boiling and Pressurized Water Reactors" (Reference 5), and for the MSLBA, Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 39). The radiological release timing methodology also incorporated the results of the BWROG Report "Prediction of the Onset of Fission Gas Release from Fuel in Generic BWR" (Reference 6) as adopted by RG 1.183.

The analysis results were evaluated to demonstrate compliance with the acceptance criteria presented in 10CFR50.67 and GDC 19 of 10CFR50 Appendix A.

UFSAR Section 15.7.2 addressed the impact of the TID-14844 source term on ESF systems, specifically, the Standby Gas Treatment System (SBGT) performance and post-accident EQ impact on the Emergency Core Cooling System (ECCS). The AST implementation impact on the SBGT performance was evaluated. The Control Room Ventilation Standby Filter Units (SFU) and Technical Support Center Filtration system were also evaluated.

2.1.2 Equipment Qualification

The scope of work for equipment qualification is based on the interim guidance of SECY-99-240 and adopted in RG 1.183 pending NRC issuance of a Generic Safety Issue on the consequences of increased radiological source term cesium isotopes in reactor coolant. The work consisted of a qualitative evaluation comparing bounding DBA (LOCA) source terms based on the TID-14844 methods for current licensed power and for operation at 1912 MWt as assumed in this report.

The AST source term was used for EQ evaluations where the predominant accident dose source comes from exposure to the radiological plume (Control Room, Control Building HVAC, pump house, and river intake structure).

2.1.3 NUREG-0737

Qualitative evaluations were performed to identify potential impacts of applying AST methodologies on compliance with NUREG-0737. Calculations were performed to update the mission doses for NUREG-0737 Item II.B.2, Shield Design and Vital Area Access Mission. Personnel doses were recalculated for evaluation of habitability for the on-site Technical Support Center (TSC) (NUREG-0737 Item III.A.1.2) and the Control Room (CR) (NUREG-0737 Item III.A.1.2).

The current DAEC design and licensing basis for Control Room Habitability in case of Toxic Gas release does not credit any control room isolation or reduction in intake flow rates. Control room inleakage values assumed in the radiological analysis are much smaller than the difference between normal flowrates and isolation flowrates. Therefore, new analysis of toxic gas effects to evaluate changes in inleakage was not performed.

This portion of the evaluation reviewed the current design basis for NUREG-0737, Item II.B.2, post-accident vital area access and determines the impact from AST implementation. Calculations used in support of NUREG-0737, Item II.B.2, shield design and vital area access mission dose assessments were included in the review. Affected documents were reviewed to determine if the current design basis contains sufficient margin to encompass the AST. Any design basis documents requiring updates in support of the AST were identified and revised if appropriate.

2.1.4 Control of Iodine Re-evolution

A calculation was performed to evaluate the pH of the Torus Suppression Pool in the event of a Design Basis Loss of Coolant Accident. The objective of this calculation was to demonstrate that, in the event of a Design Basis Loss of Coolant Accident, the pH of the Torus Suppression Pool remains above 7.0; thus ensuring that the aerosol particulate iodine (Cesium Iodide, CsI) deposited into the suppression pool during this event does not re-evolve and become airborne as elemental iodine. The analysis credits the pH buffering effect of sodium pentaborate injection with the SLCS.

2.2 Method of Evaluation

2.2.1 Fission Product Inventory

The ORIGEN2 code (Reference 1), which is a widely used Oak Ridge National Laboratory code used in the production and decay of radioactive material, was used in the calculation of plant-specific fission product inventories which bound the effect of two year fuel cycles, power operation at 1950 MWt (102% of 1912 MWt), and anticipated fuel designs.

2.2.2 Accident Radiological Consequence Analyses

New calculations were prepared for the simulation of the radionuclide release, transport, removal, and dose estimates associated with the plant's postulated accidents as listed in Section 2.1.1.

The MicroShield code, version 5.03a (Reference 13), was used in this task. MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. This version of the code has been verified and validated in accordance with DAEC Administrative Control Procedure (ACP) 102.14, "Software Quality Assurance Program." Although not considered an NRC approved code, MicroShield was used in safety related applications by many nuclear plants in the United States, including DAEC. The code has been used to support licensing submittals that have been accepted by the NRC.

The RADTRAD computer code (Reference 7), version 3.02, was used in this task. RADTRAD is a radiological consequence analysis code used to estimate post-accident doses at plant offsite locations and in the control room. This version of the code was

verified and validated in accordance with DAEC Administrative Control Procedure (ACP) 102.14, "Software Quality Assurance Program." The code is used by the NRC staff in safety reviews.

Offsite atmospheric dispersion factors (χ/Q's) were calculated with the PAVAN computer code (Reference 9). The PAVAN code calculates the diffusion from a source and relative concentration at a receiver due to an accidental release of radioactivity into the environment per the guidance in USNRC RG 1.145 (Reference 14). The PAVAN code was verified and validated in accordance with DAEC Administrative Control Procedure (ACP) 102.14, "Software Quality Assurance Program." The code is used by the NRC staff in safety reviews.

Control room atmospheric dispersion factors (χ/Q's) were calculated with the ARCON96 computer code (Reference 8). The ARCON96 code calculates relative concentrations in plumes from nuclear power plants at control room air intakes in the vicinity of the release point. The ARCON96 code was verified and validated in accordance with DAEC Administrative Control Procedure (ACP) 102.14, "Software Quality Assurance Program." The code is used by the NRC staff in safety reviews.

In addition to the calculational tools described above, the radiological consequence analyses make use of hand calculations and spreadsheets, supported by appropriate references, to determine inputs and outputs such as plant specific source terms, filter loading determinations, and suppression pool pH analyses.

2.2.3 NUREG-0737, Item II.B.2

Plant calculations used in support of the plant post-accident vital area access in accordance with the requirements of NUREG-0737, Item II.B.2, were reviewed. Post-accident personnel missions resulting in the bounding mission doses were identified. These mission doses were then reviewed for impact from the AST. Calculations were prepared to document the AST impact.

The MicroShield code, version 5.03a (Reference 13), was used in this task. MicroShield is a point kernel integration code used for general purpose gamma shielding analysis. This version of the code has been verified and validated in accordance with DAEC Administrative Control Procedure (ACP) 102.14, "Software Quality Assurance Program."

2.2.4 pH Control Calculations

The calculation methodology is based on the approach outlined in References 3, 5, and 42 (primarily Reference 42). DAEC design inputs were conservatively established to maximize the post-LOCA production of acids (Hydriodic, HI; Nitric, HNO_3; and Hydrochloric, HCl) and minimize the post-LOCA production and/or addition of bases (Cesium Hydroxide, CsOH and Sodium Pentaborate, $Na_2O*5B_2O_3*10H_2O$). Other design input values such as SP volume and initial pH are conservatively selected to minimize the calculated pH.

2.3 Inputs and Assumptions

2.3.1 Alternative Radiological Source Term

Plant-specific 24-month fuel cycle design parameters were used to calculate an estimated average effective full power days (EFPD) for DAEC. Table 2-1 summarizes key cycle parameters

The base case was provided by General Electric and consisted of a pre-existing analysis of fission product inventories from higher burnups associated with 24 month cycles. The base case used a bounding value of exposure of 1500 EFPD which exceeds the DAEC cycle value of 1132 EFPD. The base case used an average enrichment value of 4.2 wt% which exceeds the predicted DAEC average enrichment of 4.15 wt%. A core average burnup of 55 GWD/MTU was assumed which exceeds the DAEC predicted average burnup of 45 GWD/MTU. The base case assumed GE12 fuel and a core average bundle power of 4.78 MW/bundle which compares to DAEC plans to load GE14 fuel with a core average bundle power of 5.19 MW/bundle. The effect of the different fuel type and higher bundle power are offset by the conservative nature of the other inputs and produce a bounding fission product inventory. This base case inventory, based on core average input parameters, was used for the LOCA and Main Steam Line Break DBAs.

To ensure that the results of Fuel Handling Accident and Control Rod Drop Accident were not biased use of core average inputs, a second inventory was provided that is based on a combination of maximum activities from the 1500 EFPD inventory and an inventory based on a mean discharge exposure of 45.03 GWD/MT, and a maximum bundle enrichment of 4.39 wt % in uranium 235. This inventory was used in the Fuel Handling Accident (FHA) and the Control Rod Drop Accident (CRDA).

The ORIGEN2 extended-burnup library designated BWRUE [Ref. 13] was used in the calculation.

The release source term was developed using the 60 isotope subset recommended by the NRC (per the RADTRAD code, Reference 7. BWR default inventory) and the guidance in NUREG-1465 (Reference 3) for release fractions, form, and timing as well as the BWROG gap release timing criteria specified in Reference 6. This postulated LOCA source term represents a change in the DAEC design and licensing bases for radiological consequence analysis.

2.3.2 Atmospheric Dispersion Factors Meteorology

Atmospheric dispersion coefficients were calculated, for each identified release path, based on site specific meteorology data collected between January 1997 and December 1999 (Reference 22) as documented in CAL-R00-PUP-001 and CAL-R00-PUP-002. The newly calculated atmospheric dispersion coefficients are more conservative than those documented in the calculations of record for their respective release/receptor points.

2.3.3 Accident Radiological Consequence Analyses

All analyses were performed at 1950 MWt (102% of 1912 MWt) in accordance with USNRC Regulatory Guide 1.49 (Reference 38).

2.3.3.1 LOCA Inputs and Assumptions

The key inputs used in this analysis were included in Tables 2-1 through 2-4. These inputs and assumptions fall into three main categories, Radionuclide Release Inputs, Radionuclide Transport Inputs, and Radionuclide Removal Inputs.

2.3.3.1.1 LOCA Release Inputs

Per plant Technical Specifications, a 2%/day primary to secondary containment leakage rate (Reference 16) was assumed. This assumption is the same used in current design and licensing basis analyses. ESF systems were assumed to leak at a rate of 1.5 gpm (Reference 16) into the Reactor Building based on conservative estimates that are 15 times higher as compared to the plant leakage surveillance procedure limits. Consideration of continuous ESF leakage is a change to previous design and licensing

bases. Both leakage rates were assumed throughout the 30-day duration of the postulated accident.

A plant Technical Specifications total allowable drywell MSIV leakage of 200 scfh (Reference 20) was assumed to leak directly into the environment via the main steam lines and the condenser during the first 24 hours of the accident. This value was assumed to decrease to 100 scfh after 24 hours per the guidelines in Appendix A, Section 6.2 of Reference 5. This MSIV leakage reduction at 24 hours constitutes a change in the basis of the radiological consequence analysis of the LOCA event for DAEC.

2.3.3.1.2 pH Control

The NUREG-1465 (Reference 3) accident isotopic release specification allows credit for deposition of iodine in the suppression pool. Once deposited, the iodine will remain in solution as long as the pool pH is maintained above 7.

Upon detection of symptoms indicating that core damage is occurring, DAEC procedures shift over to severe accident scenarios. Operators are directed to manually initiate the Standby Liquid Control System (SBLC) upon initiation of severe accident procedures. No credit was taken for any operator action during the first 10 minutes of an event. If an accident were to occur which would create the conditions assumed in the analyses, it is reasonable to assume that manual initiation of SBLC injection would be initiated promptly. For the purposes of this analysis, however, this action was delayed for approximately 24 minutes (Reference 17) after the event starts (or essentially within 22 minutes following the onset of fission product release). This will result in complete injection of the system's 2500 gallon inventory (Reference 17) of sodium pentaborate solution into the RPV within 2 hours after the postulated accident.

Table 2-2 contains a number of the inputs used for this evaluation. The buffering effect of the SLCS solution was shown to maintain the suppression pool pH above 7 for the 30-day duration of the postulated LOCA. Calculation CAL-R00-PUP-003 documents this calculation. This is sufficient to prevent significant iodine re-evolution. As indicated above, use of the SLCS is already a DAEC standard severe accident management procedure when core damage is believed to be occurring, however, crediting this system as an assumption in support of radiological consequences analysis represents a change to the DAEC design and licensing basis.

The following list summarizes significant assumptions used in the analysis:

1. Core concrete aerosols are basic materials produced from the interaction of molten core materials with the concrete primary containment. For DBA LOCAs core damage is assumed to be arrested after the early in-vessel release phase. Thus, these chemicals were not considered within this assessment.

2. Minimum SP volume to be utilized for the post-LOCA assessment of SP pH is 58,900 ft^3. The maximum SP volume, without the relatively small Sodium Pentaborate addition from the SLCS, to be utilized for the post-LOCA assessment of SP pH is 68,312 ft^3 [61,500 + 6187 + 625].

3. SLCS data assumed a volume of 2500 gallons of 11.8 wt% $Na_2O*5B_2O_3*10H_2O$ injected at 26 gpm within two hours of the onset of the accident.

4. In approximately 1-minute post DBA LOCA start, ECCS Core Spray & LPCI pumps are available to draw suction from the Torus SP. At least 7038 gpm (1-LPCS pump at 2718 gpm and 1-RHR pump at 4320 gpm, will be circulating from the SP to the reactor vessel and/or spray system and spilling back into the SP via the vent/downcomer system. Based on the pool and coolant inventory of 68,312 ft^3, this ECCS flow represents approximately one complete exchange of the volume per 1.2 hours. On this basis, complete mixing is assumed such that a single SP pH can be applied.

5. The allowable SP pH range is 6.8 to 7.30. The allowable reactor coolant pH ranges from 5.6 to 8.6. To conservatively simplify this SP pH assessment, the initial SP pH will be conservatively considered at the minimum reactor coolant pH value so that no SP pH change need be considered due to the released reactor coolant mixing with the SP inventory. The minimum reactor coolant pH of 5.6 specified is obtained from a depressurized reactor coolant sample.

6. The cesium that is not in the chemical form of CsI is assumed to exit the RCS in the form of cesium hydroxide (CsOH) and be deposited into the SP. The rate of modeled CsOH deposition into the SP could be reasonably based on the 10% Powers model for natural deposition of aerosols as utilized within the RADTRAD computer code. However, since the Powers model only addresses

deposition from the primary containment vapor region, simplifying assumptions are made for CsOH deposition. Two cases were evaluated. For Case 1, all CsOH was assumed to reach the SP. In the resulting calculation, SP pH fell below 7.0. This shows that the effect of CsOH alone is insufficient to maintain a basic pH. For Case 2 (addition of Sodium Pentaborate), conservatively, no credit for CsOH deposition into the SP is taken.

7. The 30-Day Integrated Dose for the DAEC Suppression Pool water, the Primary Containment (DW) air gamma plus bremsstrahlung, and the DW air beta were developed. These integrated dose values are based on TID-14844 source terms at the DAEC with an assumed power level of 1950 MWt (102% of 1912 MWt). These integrated dose profiles were curve fit for inclusion in the pH calculations. Comparison of the curve fit dose values versus the reference dose values given in drywell EQ dose calculations are quite good. To conservatively ensure that the curve fit dose value at any time point greater than 0.53661-hours (0.5 hour + 121 seconds is conservatively bounded, a margin factor of 1% is applied to all curve fit integrated dose values.

8. It is expected that the initial effects on pool pH will come from rapid fission product transport and formation of CsOH. As radiolytic production of nitric acid and hydrochloric acid proceeds and these acids are transported to the pool over the first hours of the event the pH would become more acidic. The buffering effect of SLCS injection within two hours is assumed to be sufficient to offset the effects of these acids as they are transported to the pool. As stated in assumption #6, the CsOH is not credited for long term pool pH. This assumption is consistent with previous NRC staff conclusions that for the first two hours of a DBA, the iodine source term behavior and its transport within the drywell will be independent of iodine reevolution and pH control.

9. The cable mass in the DAEC DW was identified from the DAEC cable and raceway database as ~23,000 lbm excluding the copper conductor mass. Conservatively, all 23,000 lbm is assumed to be chloride-bearing.

10. Essentially the only free hanging cable within the DAEC DW is that associated with the loops at the bottom of the vessel and entering junction boxes. This analysis conservatively assumed 50% of the identified cable mass as being "free drop, 11,500 lbm. The other half of the chloride-bearing cable

mass (non "free drop" cable mass) is assumed to experience a beta radiation dose equal to 50% of the incident dose due to self and structural shielding.

11. From the DAEC cable information the largest cable radius of 2.22 inches was utilized to conservatively maximize the absorption fraction of the incident gamma and beta energy within the chloride-bearing cables, thus maximizing the Hydrochloric Acid generated.

12. The production of all the strong acids and bases except Nitric Acid are independent of SP volume. The equations utilized for these acids and bases simply divides the production value by the SP volume to obtain the acid or base concentration within the SP. Nitric Acid production is a direct function of the SP volume; however, as pool volume increases, the specific radiation activity decreases proportionally since the total radiation source term does not change. Thus for Nitric Acid production, the specific radiation dose in a liter of the SP drops proportionally with the increase in SP volume. Therefore, the production of all acids and bases is independent of SP volume. The calculation case studies will be performed using the maximum pool volume (and minimum SP initial pH) since the maximum SP volume will be the bounding case when considering the addition of the Sodium Pentaborate buffer (greater dilution of the buffer).

13. When calculating the average to incident beta radiation flux for the production of Hydrochloric Acid in chloride-bearing cable, it was assumed that the jacket thickness of all cables is 0.045 Inches. This assumption conservatively bounds DAEC plant configuration and results in greater Hydrochloric Acid production. Also, minimum jacket thickness and the minimum cable radius values were assumed to maximize acid production from these cables. Similarly, when calculating the Hypalon mass to cable mass ratio used in the calculation of Hydrochloric Acid from beta radiation, the jacket thickness value used for all cables is the maximum value. This assumption conservatively maximizes acid production from this term.

2.3.3.1.3 LOCA Transport Inputs

At the beginning of the event, a loss of offsite power was assumed which results in the loss of reactor building ventilation that maintains secondary containment at a

negative pressure with respect to the outside atmosphere. A Positive Pressure Period (PPP) was conservatively assumed to occur at time zero resulting in leakage from the secondary containment to the environment. The PPP was conservatively assumed to be terminated at 5 minutes including time delays for Emergency Diesel Generators (EDG) start and the SBGT system start and secondary containment negative pressure drawdown to 0.25 inHg vacuum. This 5-minute PPP was chosen to be sufficiently conservative such that it would bound actual system performance. Addition of this assumption to the current basis is necessary for compliance with Appendix A of RG 1.183, Reference 39. During the PPP, both the Containment and ESF leakage were assumed released directly to the environment, unfiltered. at ground level. To maximize the calculated post-accident doses, the ground level Reactor Building releases during the PPP were assumed to discharge from the building's closest location to the receptor location at the Control Room (CR) or the Technical Support Center (TSC) air intake. Both locations correspond to points along the Reactor Building's North wall (References 22, 23, and 28). Following the PPP, all RB releases discharge from the SBGT stack.

The MSIV leak pathway was assumed to discharge from the condenser at approximately the center of the Turbine Building's ground elevation (References 22, 23, and 28). This assumption implies dispersion of the activity at this "average" location corresponding to the building's available openings with no credit for building hold-up.

The CR is automatically isolated and placed in the Emergency ventilation mode upon an air intake radiation monitor isolation signal. For analysis purposes, the accident activity was allowed to enter the CR for up to four (4) minutes (Reference 24) into the LOCA at the normal ventilation flow rate of 3150 cfm (Reference 24). The assumed configuration of the CR ventilation and isolation is consistent with the delayed nature of the AST and differs from the current design basis that assumes instantaneous CR isolation based on an instantaneous source term release. Sensitivity analyses of the post-LOCA dose to CR operators were performed at assumed unfiltered CR inleakage rates of 67.5, 500, and 1000 cfm (Reference 17). An additional sensitivity case was performed assuming 0 cfm in leakage which resulted in a very small (0.01 Rem TEDE) increase over the Reference 17 results at 67.5 cfm.

The TSC is isolated by operator action. A conservative operator isolation at 30 minutes was assumed (Reference 16). Similar to the CR isolation, this delayed TSC isolation is consistent with the nature of the AST release and differs from the current design basis that assumes instantaneous TSC isolation based on an instantaneous source

term release. Prior to isolation, the TSC normal ventilation flow rate is 900 cfm (Reference 16). Sensitivity analyses of the post-LOCA dose to the TSC were performed at assumed unfiltered TSC inleakage rates of 67.5, 500, and 1000 cfm (Reference 17).

2.3.3.1.4 LOCA Removal Inputs

The accident's activity released from the core is partially removed by natural deposition mechanisms in the drywell, Main Steam lines, and Condenser as well as by air filtration systems in the RB, CR, and TSC. The natural deposition removal mechanisms are characteristic of the nature of the AST and represent a change in the plant design and licensing basis.

Drywell natural deposition was simulated using the 10th percentile data for the Power's natural deposition model in the RADTRAD code (Reference 7).

Main steam line pipe deposition was simulated using the RADTRAD code's (Reference 7) Brockmann – Bixler pipe deposition model using plant specific piping geometry data.

Activity deposition in the plant condenser was estimated using the DAEC condenser deposition filter efficiency calculated by GENE in References 10 and 25.

Filter removal at the SBGT and the Emergency modes of the CR and TSC Ventilation systems was simulated using plant design data as listed in Table 2-3. The SBGT Bypass Flow assumed is the maximum allowed by plant Technical Specifications.

Conservatively an aerosol removal efficiency of 99% (References 16 and 17) was used for the SBGT, SFU and TSC ventilation system HEPA filters. The HEPA filter efficiencies used are the same as those utilized in the existing licensing basis analyses. The SBGT charcoal filter efficiency of 99% for elemental and organic iodines (Reference 16) is consistent with Regulatory Guide 1.52 (Reference 18) for a 6 inch charcoal bed with humidity control. The SBGT charcoal filter efficiencies used are the same as those utilized in the current licensing analyses. The CR and TSC charcoal filter efficiencies of 90% for elemental iodine and 30% for organic iodine (Reference 16) are consistent with Regulatory Guide 1.52 for a 2 inch charcoal bed without humidity control. The CR and TSC charcoal filter efficiencies used are more conservative than those utilized in the current licensing analyses.

2.3.3.1.5 Personnel Dose Conversion Inputs

The standard breathing rate of 3.47E-4 m^3/sec for the 30-day accident duration (References 5 and 7) was used for onsite personnel in the CR and TSC dose assessments. Likewise, the standard offsite breathing rates of 3.47E-04 m^3/sec from 0 to 8-hours, 1.75E-04 m^3/sec from 8 to 24-hours, and 2.32E-04 m^3/sec from 1 to 30-days is used in dose assessments. Occupancy factors in the CR and TSC were 1.0 from 0 to 1-day, 0.6 from 1 to 4-days, and 0.4 for 4 to 30-days. The values are the same as those utilized in the existing licensing analyses.

CR operator ingress and egress dose was evaluated based on an operator performing 22 shifts (Reference 19) during the 30-days post accident. The operator was assumed to be in the CR during the first 24-period (3 shifts)(Reference 19) and was conservatively assumed to report daily for the next nineteen days; thus making 39 ingress/egress trips (Reference 19) during the accident. The CR operators were assumed to not use any protective clothing or breathing apparatus (Reference 19). The ingress/egress pathway was assumed from the site boundary to the guard facility through the Administration Building to the CR with a total travel length of 6.4 minutes (Reference 19). These assumptions are more conservative than those utilized in the existing licensing analyses.

These key inputs are summarized in Table 2-4.

2.3.3.2 MSLBA Inputs and Assumptions

The key inputs used in this analysis are included in Table 2-5.

The postulated accident assumes a double ended break (DEB) of one main steam line, with the reactor operating at 1950 MWt (102% of 1912 MWt), outside the secondary containment with displacement of the pipe ends that permits maximum blowdown rates.

The break mass released includes the line inventory plus the system mass released through the break prior to isolation. Break isolation was assumed in 10.5 seconds (Reference 26). This assumption is consistent with the isolation time used in evaluation of the HELB pressure, temperature, pipe whip and jet impingement for MSLBs. It is a conservatively longer isolation time than the expected 3 to 5 second isolation based on MSIV technical specifications and testing results. This also resulted in a conservatively large radiological release for analysis.

Two activity release cases corresponding to the maximum pre-accident spike and maximum equilibrium concentration allowed by proposed changes to plant Technical Specifications. 2 μCi/gm (pre-accident spike) and 0.2 μCi/gm (maximum equilibrium concentration) dose equivalent I-131 respectively were assumed. These released activity assumptions are consistent with the requirements of RG 1.183, Reference 39.

All the accident activity was assumed released instantaneously (i.e., within 10.5 seconds corresponding to the break duration) from the Turbine Building as a ground release without credit for building holdup or dilution (Reference 39).

The CR was conservatively assumed to automatically isolate in 10.5 seconds following the accident. This isolation timing was determined by analysis to be the most conservative and maximizes the CR dose. This CR isolation assumption is consistent with the nature of the release and differs from the current design basis that assumes instantaneous CR isolation based on an instantaneous source term release. Prior to isolation, the activity is assumed to enter the control room at the normal ventilation rate of 3150 cfm (Reference 24). Sensitivity analyses of the post-MSLB dose to CR operators were performed at assumed unfiltered CR inleakage rates of 0, 67.5, 500, and 1000 cfm (Reference 17).

Filter removal by the Emergency modes of the CR Ventilation system was simulated using plant design data as listed in Table 2-3 as well as cases in which the CR Emergency Mode Ventilation flow rate was assumed at 900 cfm per the low limit in plant Technical Specification 3.7.4 (Reference 40).

2.3.3.3 Refueling Accident Inputs and Assumptions

The key inputs used in this analysis are included in Table 2-6.

This postulated Fuel Handling Accident (FHA) involves the drop of a fuel assembly on top of the reactor core during refueling operations. The drop over the reactor core is more limiting than the drop over the spent fuel pool because the kinetic energy for the drop of thirty feet over the vessel produces a much greater number of damaged fuel pins on impact than the shorter drops that could occur over the fuel pool (Reference 17). Based on limiting considerations, 151 fuel rods (References 29 and 30) were assumed damaged. Per Reference 37, a radial peaking factor of 1.46 was assumed. The assumed number of rods damaged is more conservative than the existing design and licensing basis value of 125 rods. Per plant refueling procedures, a post-shutdown 60-hour decay period

(Reference 17) was used to determine the release activity inventory. This decay period constitutes an increase over the existing design and licensing basis of 24 hours. All the gap activity in the affected rods was assumed released instantaneously into the pool. A pool water iodine decontamination factor of 200 was used based on the guidelines of Reference 39. This pool water decontamination factor is higher than the value of 100 used in the existing licensing basis analysis.

Conservatively, all the FHA activity was assumed released within two hours (Reference 5) from the Reactor Building as a ground release (Reference 22, 23, and 28) with no credit for Containment and/or Reactor Building holdup, Containment and/or Reactor Building dilution, or SBGT operation. The position of not crediting any dilution, holdup, or cleanup by the SBGT system of the activity released from the pool represents a more conservative basis than that used in the existing licensing basis analysis.

2.3.3.4 Control Rod Drop Accident Inputs and Assumptions

The key inputs used in this analysis are included in Table 2-7.

The plant design basis Control Rod Drop Accident (CRDA) involves the rapid removal of a high worth control rod resulting in a reactivity excursion that encompasses the consequences of any other postulated CRDA. Based on limiting assumptions, a total of 1200 fuel rods (Reference 32) were assumed damaged with 0.77 % (Reference 33) of the damaged rods clad melting. Per Reference 37, a radial peaking factor of 1.46 was assumed. The assumed number of rods damaged is more conservative than the existing design and licensing basis value of 850 rods.

A fraction of the activity released from the damaged fuel was assumed to reach the turbine condenser where it is available to be released from the Turbine Building at ground level (References 22, 23, and 28) at a rate of 1% per day for a period of 24 hours. No credit was taken for Turbine Building holdup or dilution (Reference 5).

2.3.4 NUREG-0737, Item II.B.2

The key inputs used in this analysis include:

- Plant-specific Fission products inventories and reactor coolant source terms.
- Post-accident cloud dose as addressed in Section 3.1.1.1 of this report.

- DAEC NUREG-0737, Item II.B.2, plant shielding and mission dose assessment reports (References 11 and 12). These reports evaluate post-accident limiting mission doses associated with Post-Accident Sampling System (PASS) sample collection and the Reactor Building Exhaust Stack sampling.

Table 2-1: 24-Month Fuel Cycle Data

24-Month Fuel Cycle Data	Base Case (BC) Used in ORIGEN2 Calculation of Fission Product Inventory	DAEC Cycle
Fuel Type	GE12	GE14
Total EFPD	1500	1132
Core Average Discharge Burnup (GWD/MT)	55	45.03
Initial Bundle Mass of Uranium (kg)	180	179.48
Initial Core Average Enrichment U-235 wt%	4.2	4.15
Core Average Bundle Power (MW/bundle)	4.78	5.19

Table 2-2: Key LOCA Analysis Inputs and Assumptions, Release Inputs

Release Inputs	
Input/Assumption	**Value**
Fission Products Core Inventory	Plant Specific ORIGEN2
Fission Products Release Fractions, Form, and Timing	RG 1.183 (Reference 39)
Primary Containment Leakage Rate	2%/day (Reference 16)
ESF Systems Leakage Rate	1.5 gpm (Reference 16)
Total MSIV Leakage	200 scfh for 24 hrs, 100 scfh afterwards (References 20 and 5)
Start of SLCS Injection	24 minutes into the LOCA (Reference 17)
SLCS Inventory	2500 gallons (Reference 21)
SLCS Injection Flow Rate	26.2 gpm (Reference 21)
SLCS Solution Composition	11.8 weight % Sodium Pentaborate (Reference 21)

Table 2-3: Key LOCA Analysis Inputs and Assumptions, Transport Inputs

Transport Inputs	
Input/Assumption	**Value**
Reactor Building PPP	5 minutes
Reactor Building Ground PPP Release Location	Closest Point to CR and TSC Intake from Reactor Building North Wall. (References 22, 23, and 28)
Turbine Building Ground MSIV Release Location	Center of Turbine Building (References 22, 23, and 28)
Meteorology	Site Specific, 01/97 to 12/99 (Reference 22)
Control Room Isolation	4 minutes (Reference 24)
CR Normal Mode Ventilation Flow Rate	3150 cfm (Reference 24)
CR Isolation Filtered Flow Rate	1000 cfm
Control Room Unfiltered Inleakage Rate	0, 67.5, 500, and 1000 cfm
Technical Support Center Isolation	30 minutes (Reference 16)
TSC Normal Mode Ventilation Flow Rate	900 cfm (Reference 16)
TSC Isolation Filtered Flow Rate	1000 cfm (200 cfm fresh, 800 cfm recirculation) (Reference 23)

Table 2-4: Key LOCA Analysis Inputs and Assumptions, Removal Inputs

Removal Inputs	
Input/Assumption	**Value**
Drywell Natural Deposition	Power's 10th Percentile Model (Reference 5)
Main Steam Lines Deposition	Brockmann - Bixler Pipe Deposition Model (Reference 5)
Condenser Deposition	GENE Condenser Deposition Model (References 10 and 25)
Elemental & Particulate Filter Efficiency	99.6%, Main Stop Valve Drain Path
Elemental & Particulate Filter Efficiency	99.7%, MSIV Drain Path
SBGT Flow Rate	4000 cfm (Reference 16)
SBGT Bypass Flow	0.1% (Reference 16)
SBGT Filter Efficiency (All species)	99% (References 16 and 17)
CR Emergency Mode Ventilation Flow Rate	1000 cfm (Reference 16)
TSC Emergency Mode Ventilation Flow Rate	200 cfm (Reference 16)
TSC Recirculation Mode Ventilation Flow Rate	800 cfm (Reference 16)
CR & TSC Emergency Mode Filter Iodine Efficiency	99%, Aerosol (References 16 and 17) 90%, Elemental (Reference 18) 30%, Organic (Reference 18)

Table 2-5: Key LOCA Analysis Inputs and Assumptions, Personnel Dose Conversion Inputs

Personnel Dose Conversion Inputs	
Input/Assumption	**Value**
Onsite Breathing Rate	3.47E-04 m^3/sec (References 5 and 7)
Offsite Breathing Rate	0-8 hours: 3.47E-04 m^3/sec 8-24 hours: 1.75E-04 m^3/sec 1-30 days: 2.32E-04 m^3/sec (References 5 and 7)
CR & TSC Occupancy Factors	0-1 day: 1.0 1-4 days: 0.6 4-30 days: 0.4 (References 5 and 7)
CR Operator Shifts during Accident	22 total, with 3 in first 24 hours (Reference 19)
CR Operator – Number of Ingress/Egress Trips	39 (Reference 19)
Ingress/Egress Transit Time	6.4 minutes (Reference 19)
CR Operator – Protective Clothing or Breathing Apparatus	None (Reference 19)

Table 2-6: Key MSLBA Analysis Inputs and Assumptions

MSLBA Inputs	
Input/Assumption	**Value**
Mass Release Rate	Same as existing design basis. (Reference 26)
Break Isolation Time	Assumed 10.5 seconds
Maximum Pre-Accident Spike Iodine Concentration	Per TS value of 2 μCi/gm (proposed change to Reference 27)
Maximum Equlibrium Iodine Concentration	Per TS value of 0.2 μCi/gm (proposed change to Reference 27)
Release Period	Assumed 10.5 seconds
Turbine Building Ground Release Location	Ground Release Turbine Building Exhaust Vent (Reference 17)
Meteorology	Site Specific, 01/97 to 12/99 (Reference 22)
Control Room Isolation	10.5 seconds (Conservative assumption)
CR Normal Mode Ventilation Flow Rate	3150 cfm (Reference 24)
Control Room Unfiltered Inleakage Rate	0, 67.5, 500, and 1000 cfm (Reference 17)

Table 2-7: Key FHA Analysis Inputs and Assumptions

FHA Inputs	
Input/Assumption	**Value**
Number of Failed Rods	151 (References 29 and 30)
Radial Peaking Factor	1.46 (Reference 37)
Fuel Decay Period	60 hours (Reference 17)
Pool Water Iodine Decontamination Factor	200 (Reference 39)
Release Period	2 hours (Reference 5)
Reactor Building Ground Release Location	Reactor Building (References 22, 23, and 28)
Meteorology	Site Specific, 01/97 to 12/99 (Reference 22)

Table 2-8: Key CRDA Analysis Inputs and Assumptions

CRDA Inputs	
Input/Assumption	**Value**
Number of Failed Rods	1200 (Reference 32)
% Fuel Melt	0.77% (Reference 33)
Radial Peaking Factor	1.46 (Reference 37)
Turbine Building Leakage Rate	1%/day (Reference 5)
Release Period	24 hours (Reference 5)
Turbine Building Ground Release Location	Turbine Building (Reference 22, 23, and 28)
Meteorology	Site Specific, 01/97 to 12/99 (Reference 22)

3. RESULTS

3.1 Evaluation Results

3.1.1 Accident Radiological Consequence Analyses

The postulated accident radiological consequence analyses were updated for AST implementation impact. Comparison of updated AST doses to existing licensing basis doses indicates impact from the assumed operation at 1912 MWt as well as the change in analysis methodology per the AST.

3.1.1.1 LOCA

The radiological consequences of the design basis LOCA were analyzed using the RADTRAD code (Reference 7) and the inputs / assumptions defined in Section 2.3.1.1 of this report. The detailed analysis was documented in CAL-R00-PUP-007. The post accident doses are the result of three distinct activity releases:

1. Primary to secondary containment leakage. This leakage is directly released into the environment during the secondary containment's positive pressure period (PPP) and filtered by the SBGT afterwards.

2. ESF system leakage into the secondary containment. This leakage is also directly released into the environment during the secondary containment's PPP and filtered by the SBGT afterwards.

3. MSIV leakage from the primary containment into the Main Condenser. This leakage is released, undiluted and unfiltered, through the Turbine Building.

Table 3-1 presents the results of the LOCA radiological consequence analysis for offsite receptors. As indicated, the Exclusion Area Boundary (EAB), 936 meters from the plant stack, and the Low Population Zone (LPZ), 2 miles from the site, (Reference 22, 23, and 28) calculated doses are within the regulatory limits after AST implementation.

The results of the LOCA radiological consequence analysis for the CR and TSC are presented in Table 3-2. The dose to both the CR and TSC occupants includes terms for:

1. In-leakage internal cloud immersion and inhalation contribution from the primary containment, ESF, and MSIV leakage releases.

2. External cloud contribution from the primary containment, ESF, and MSIV leakage releases. This term takes credit for CR/TSC structural shielding.

3. A direct dose contribution from the Reactor Building contained accident activity. This term takes credit for both Reactor Building and CR/TSC structural shielding.

In addition, the total CR dose includes a RB/CR shielded contribution from the SBGT filters as well as the occupants' dose associated with shift ingress/egress missions during the 30-day post accident period.

Per Table 3-2, the post LOCA CR and TSC calculated doses are within regulatory limits after AST implementation.

The Table 3-2 post-LOCA CR dose corresponds to an assumed unfiltered inleakage rate of 67.5 cfm (Reference 17). The LOCA CR unfiltered inleakage sensitivity analyses demonstrated that the post-LOCA CR doses are maximized at the lower inleakage rate due to the additional CR envelope atmosphere cleaning rate offered by higher inleakage rates after the end of the positive pressure period (PPP) primary containment release.

3.1.1.2 MSLBA

The radiological consequences of the design basis MSLBA were analyzed in CAL-R00-PUP-008 using the RADTRAD code (Reference 7) and the inputs / assumptions defined in Section 2.3.1.2 of this report. Two activity release cases corresponding to the reactor coolant maximum pre-accident spike and maximum

equilibrium concentration allowed by proposed plant Technical Specifications or 2 μCi/gm and 0.2 μCi/gm dose equivalent I-131 respectively (Reference 27) were analyzed.

Table 3-3 presents the results of the MSLBA radiological consequence analysis for offsite receptors. As indicated, both the EAB and LPZ calculated doses are within the regulatory limits after AST implementation for both cases analyzed.

The calculated CR doses for the MSLBA are presented in Table 3-4. The doses are bounded by the LOCA event doses and are within the regulatory limits after AST implementation for both cases analyzed.

The Table 3-4 post-MSLBA CR dose corresponds to an assumed unfiltered inleakage rate of zero cfm. The MSLBA CR unfiltered inleakage sensitivity analyses demonstrated that the post-MSLBA CR doses are maximized at zero inleakage rate due to the instantaneous nature of the release and the "clean-up" effect of higher inleakage rates following the release.

3.1.1.3 Refueling Accident

The radiological consequences of the design basis FHA were analyzed in CAL-R00-PUP-008 using the RADTRAD code (Reference 7) and the inputs / assumptions defined in Section 2.3.1.3 of this report. Table 3-5 presents the results of the FHA radiological consequence analysis for offsite receptors. As indicated, both the EAB and LPZ calculated doses are within the regulatory limits after AST implementation. Control Room and TSC doses are also within regulatory limits.

3.1.1.4 Control Rod Drop Accident

The radiological consequences of the design basis CRDA were analyzed in CAL-R00-PUP-008 using the RADTRAD code (Reference 7) and the inputs / assumptions defined in Section 2.3.1.4 of this report. Table 3-6 presents the results of the CRDA radiological consequence analysis for offsite receptors. As indicated, both the EAB and LPZ calculated doses are within the regulatory limits after AST implementation.

3.1.1.5 Plant Area Post-Accident Radiation Levels

The post-accident radiation levels to plant areas, other than the personnel doses in the CR/TSC and NUREG-0737 post-accident mission doses addressed in this task, are relevant for EQ assessments only.

Minor changes were identified in EQ accident doses to the Control Room, Control Room HVAC Area, Intake Structure, and Pumphouse. None of these changes resulted in designation of these areas as radiologically harsh environments. The predominant source of radiation to these areas is from the plume, and therefore, as an exception to the use of the TID-14844 source term for other EQ evaluations as discussed in section 2.1.2, calculations for these areas used the AST LOCA source term.

For all other areas, calculations show that the AEP post accident EQ area dose rates and 30 day gamma integrated doses are bounded by the values used in plant EQ program design documentation.

3.1.1.6 Impact on Plant Radiation Monitors

The plant radiation monitors and setpoints evaluated for impact for operation at the proposed power of 1912 MWt and for the impact of implementing AST. Impacts due to uprated power will be addressed in the application for extended power uprate. This section addresses the impact of AST implementation on the Analytical Limits (AL) of the Reactor Building Area Exhaust High Radiation Monitors (RIM-7606A&B), the Refueling Area Exhaust High Radiation Monitors (RIS-413A&B), and the Main Steam Line High Radiation Monitor setpoints.

The design basis for both the RIM-7606A&B and RIS-413A&B setpoint's AL was based on a radiological analysis of the detector response to a postulated accident resulting in offsite doses at the regulatory limit (References 34 and 35). The analysis assumes atmospheric dispersion factors that have been updated for AST implementation per CAL-R00-PUP-001, Rev. 1," Accident Offsite Radiological Atmospheric Dispersion Factors (Chi/Q)." Table 5 of CAL-R00-PUP-001 demonstrates that the USNRC RG 1.3

based atmospheric dispersion factors used in the Reference 35 AL evaluation are conservative as compared to those used in the updated AST analyses. Therefore, the current basis for the RIM-7606A&B and RIS-413A&B setpoint's AL is not impacted by the AST implementation.

Per Reference 36, the Main Steam Line Radiation Monitor (MSLRM) AL is established not to exceed 5.7 R/hr. Selection of this AL ensures MSLRM detection of the CRDA. The calculation utilizes the activity release to the reactor vessel resulting from the CRDA event and the MS flow rate to determine the AL of 5.7 R/hr.

The activity release (into the reactor vessel) for the CRDA as given in DAEC UFSAR Section 15.4.7 and utilized within Reference 36 is:

Iodines:4.4E+04 Curies; Noble Gases:1.32E+05 Curies (Current Design Basis)

Attachments 17 & 18 of CAL-R00-PUP-008 provide the following activity release (into the reactor vessel) values for the CRDA event using the AST at the uprate power of 1950 MWt:

Iodines:2.58E+06 Curies; Noble Gases:1.234E+06 Curies

As prescribed by Appendix C of Reference 5, 10% of the iodines and 100% of the noble gases available in the reactor vessel were postulated to reach the turbine/condenser in the AST CRDA analysis. Adjusting the reactor vessel activity accordingly yields:

Iodines:2.58E+05 Curies; Noble Gases:1.234E+06 Curies (AST Analysis)

Since both the iodine and noble gas release activity postulated in the AST analysis exceed the activity values used in Reference 36 to establish the current MSLRM AL of 5.7 R/hr, it can be concluded that maintaining the current AL value for the MSLRM will ensure detection of the CRDA following AST implementation.

3.1.2 NUREG-0737, Item II.B.2

The DAEC calculations prepared in support of the NUREG-0737, Item II.B.2, plant shielding and mission dose assessment reports were reviewed. These calculations contain the licensing basis for the DAEC NUREG-0737, Item II.B.2, plant shielding and

mission dose assessment. Calculations were revised or superceded as needed to support the new mission dose assessments. Reference 12 indicates that the DAEC vital area access mission with the highest dose to personnel was associated with a mission to the Post-Accident Sampling System (PASS). Controlling vital area access missions were identified and their impact from the AST is documented in a new calculation, CAL-R00-PUP-006.

The AST impacts the source terms used in the evaluation of vital area access mission doses. Specifically, CAL-R00-PUP-006 addresses the effects on mission dose evaluations from revised post-accident cloud immersion, reactor building shine, and reactor coolant contained source doses associated with the AST. Using the controlling PASS mission dose, CAL-R00-PUP-006 demonstrates that the DAEC plant shielding is adequate to maintain the post-accident vital area access considering the impact from the AST as shown in Table 3-7.

3.1.3 Atmospheric Dispersion Factors

Tables 1 and 2 provide the results of the PAVAN-PC analysis for the maximum sector χ/Q's.

Table No. 1 – Ground Level Release χ/Q's from the Reactor Building sec/m^3

Ground Level Release χ/Q's from the Reactor Building Sec/m^3		
	EAB (629m ENE)	LPZ (3218 m NE)
0 – 2 hours	5.57-4	1.34-4
0 - 8 hours	3.42-4	6.43-5
8 – 24 hours	2.69-4	4.46-5
1 - 4 days	1.59-4	2.01-5
4 – 30 days	7.43-5	6.43-6

Table No. 2 – Elevated Release χ/Q's from the Off Gas Stack sec/m^3

Elevated Release χ/Q's from the Off Gas Stack sec/m^3		
	EAB (936m NW)	LPZ (3218 m NW)
Fumigation	7.03-5(1)	3.15-5(1)
0 – 2 hours	6.95-6	6.69-6
0 - 8 hours	3.61-6	3.58-6
8 – 24 hours	2.61-6	2.61-6
1 - 4 days	1.28-6	1.32-6
4 – 30 days	4.64-7	4.99-7

(1): The maximum fumigation values occur in the E sector; 1.18-4 for the EAB and 3.64-5 for the LPZ.

The ground level release control room χ/Q values for the TB exhaust path for over the RB case are presented in Table 3.

Table No. 3– GROUND LEVEL RELEASE TB EXHAUST χ/Q (sec/m^3)

GROUND LEVEL RELEASE χ/Q's (sec/m^3)	
Time Period	ARCON96 D=82 m A=1609 m^2 AZ=153° Sector=90°
0 – 2 hours	9.23-4
2 – 8 hours	7.96-4
8 – 24 hours	3.57-4
1 - 4 days	2.47-4-4
4 – 30 days	1.88 -47.52-5

The ground level release χ/Q's for the release from the condenser for the CR and TSC are provided in Table 4.

Table No. 4– CR and TSC GROUND LEVEL RELEASE CONDENSER χ/Q (sec/m^3)

CR and TSC GROUND LEVEL RELEASE CONDENSER χ/Q (sec/m^3)		
Time Period	CR ARCON96 D=60.8 m A=1609 m^2 AZ=137° Sector=90°	TSC ARCON96 D=52.5 m A=1609 m^2 AZ=168.5° Sector=90°
0 – 2 hours	1.48-3	2.14-3
2 – 8 hours	1.27-3	1.86-3
8 - 24 hours	5.56-4	8.44-4
1 - 4 days	3.40-4	6.10-4
4 - 30 days	2.65-4	4.69-4

The ground level release χ/Q's for the release from the closest location on the RB wall for the CR and TSC are provided in Table 5.

Table No. 5– CR and TSC GROUND LEVEL RELEASE RB WALL χ/Q (sec/m^3)

CR and TSC GROUND LEVEL RELEASE RB WALL χ/Q (sec/m^3)		
Time Period	CR ARCON96 D=15.8 m A=1609 m^2 AZ=180° Sector=90°	TSC ARCON96 D=22.6 m A=1609 m^2 AZ=192° Sector=90°
0 – 2 hours	1.33-2	8.52-3
2 – 8 hours	1.12-2	7.09-3
8 - 24 hours	5.21-3	3.28-3
1 - 4 days	3.77-3	2.36-3
4 - 30 days	2.87-3	1.86-3

The elevated release control room χ/Q values for releases from the off gas stack are presented as follows in Table 6.

Table No. 6– CR ELEVATED RELEASE χ/Q OFF GAS STACK (sec/m^3)

Time Period	ARCON96 D=210 m A=1609 m^2 AZ=165° Sector=90°	PAVAN-PC D=210 m A=1609 m^2
30 minutes (fumigation)	Not calculated	2.62-4
0 - 8 hours	3.80-7(1)	
0 – 2 hours	3.93-7(2)	
2 – 8 hours	3.75-7(2)	
8 – 24 hours	1.33-7(2)	
1 – 4 days	1.04-7	
4 – 30 days	9.37-8	

(1): Based on 0 to 8 hour averaging period.

(2): Based on ARCON96 standard averaging intervals.

The elevated release TSC χ/Q values for releases from the off gas stack are presented in Table 7.

Table No. 7– TSC ELEVATED RELEASE χ/Q OFF GAS STACK (sec/m^3)

Time Period	ARCON96 D=214 m A=1609 m^2 AZ=173° Sector=90°	PAVAN D=214 m A=1609 m^2
30 minutes (fumigation)	Not calculated	2.38-4
0 - 8 hours	2.20-7(1)	
0 – 2 hours	2.32-7(2)	
2 – 8 hours	2.16-7(2)	
8 - 24 hours	8.00-8(2)	
1 - 4 days	6.15-8	
4 - 30 days	5.39-8	

(1): Based on 0 to 8 hour averaging period.

(2): Based on ARCON96 standard averaging intervals.

Table No. 8– FHA CR/TSC GROUND RELEASE χ/Q RB VENT (sec/m^3)

Time Period	CR – RB Vent ARCON96 D=29.2 m A=1609 m^2 AZ=125° Sector=90°	TSC – RB Vent ARCON96 D=24.4 m A=1609 m^2 AZ=196° Sector=90°
0 – 2 hours	2.85-3	2.66-3
2 – 8 hours	2.29-3	2.25-3
8 - 24 hours	1.02-3	1.03-3
1 - 4 days	3.64-4	4.34-4
4 - 30 days	1.80-4	2.30-4

Based on the analyses the χ/Q values for use in the radiological analyses are given as follows:

1. For Offsite doses at the EAB and LPZ, the values used are the PAVAN values from Tables 1 and 2.

2. For releases from the turbine building exhaust vent the CR values are the ARCON96 values listed in Table 3.

3. For releases from the condenser the CR and TSC values are the ARCON96 values listed in Table 4.

4. For releases from the reactor building closest location the CR and TSC values are the ARCON96 values listed in Table 5.

5. For the FHA releases from the refueling floor to the CR and TSC, the release is treated as a ground level release at the location of the building exhaust vent. Values are listed in Table 8.

6. For releases from the off gas stack the CR and TSC values are the PAVAN-PC values for fumigation and the ARCON96 values based on the standard ARCON96 averaging intervals for the remaining time periods listed in Table 6.

3.1.4 Post-Accident Containment Water Chemistry Management

The reevolution of elemental iodine from the suppression pool is strongly dependent on pool pH. The analysis assumed that sodium pentaborate was injected via the SLCS within two hours of the onset of a DBA LOCA. The conservative modeling of the DAEC containment cabling produce a large amount of hydrochloric acid. The minimum pool pH at 30 days post-LOCA is 8.25 which satisfies the conditions for minimizing reevolution of elemental iodine. The acid production results are summarized in Table 3-8 and the pH response over time is shown in Figure 3-1.

3.2 Key Parameter Comparison

This task addresses the plant accident radiological consequence analyses for impact from AST implementation. Implementation of the AST required a complete new set of analysis inputs and assumptions consistent with the requirements of Reference 5 (Reference 39 for the MSLBA). These new inputs were not necessarily consistent with the original licensing basis analyses and thus a key parameter comparison is not applicable to this task.

3.3 Evaluation Conclusions

3.3.1 Accident Radiological Consequence Analyses

As shown in Tables 3-1 through 3-6, the plant's accident radiological consequence analyses demonstrate that the post accident offsite and CR/TSC doses can be maintained below regulatory limits following AST implementation. All analyses were performed at 102% of 1912 MWt and in accordance with Reference 5 for AST implementation (Reference 39 for MSLBA). No changes to the plant configuration are required to support any of the analyses. It should be noted that the AST implementation, specifically the AST, results in a shift in the relative impact of the postulated accidents on offsite and CR doses.

3.3.2 NUREG-0737, Item II.B.2

An evaluation of the bounding post-accident mission, the PASS sampling mission, shows that the combined effects of the AST result in a minor increase in post-accident vital area access mission doses from the cloud immersion and reactor water contained source dose components of the mission dose. However, it has been demonstrated that the current plant shielding is adequate to maintain the post-accident vital area access considering the impact from the AST.

Table 3-1: LOCA Radiological Consequence Analysis, Offsite Doses

Dose Component	**EAB[1] (rem TEDE)**	**LPZ[2] (rem TEDE)**
Primary Containment Leakage	0.201	0.395
ESF Leakage	0.003	0.015
MSIV Leakage	0.046	0.191
TOTAL	**0.250**	**0.601**
Regulatory Limit	25	25
Current Licensing Basis[3]	7.65 (25) Whole Body 28.7 (300) Thyroid	2.90 (25) Whole Body 42.5 (300) Thyroid

1. Worst 2-hour integrated dose.

2. 30-day integrated dose.

3. Whole Body and Thyroid dose in rem, current regulatory limit in parenthesis.

Table 3-2: LOCA Radiological Consequence Analysis, CR and TSC Doses

Dose Component	**CR[1] (rem TEDE)**	**TSC[2] (rem TEDE)**
Primary Containment Leakage	2.303	1.149
ESF Leakage	0.041	0.021
MSIV Leakage	0.789	1.613
External Cloud	0.002	0.104
RB Direct Shine	0.094	1.510
SBGT Filter Direct Shine	0.049	N/A
Ingress/Egress	0.874	N/A
TOTAL	**4.152**	**4.397**
Regulatory Limit	5	5
Current Licensing Basis[3]	0.65 (5) Whole Body 22.6 (30) Thyroid 9.0 (30) Skin	4.78 (5) Whole Body 8.15 (30) Thyroid 8.9 (30) Skin

1. Assumes unfiltered inleakage of 67.5 cfm (Reference 17). For 0 cfm inleakage total dose sensitivity case Total CR dose increased to 4.16 R TEDE. For CR LOCA dose calculations, the primary containment release during the PPP is dominant. The activity for this release terminates at 5 minutes into the accident. The LOCA CR unfiltered inleakage sensitivity analyses demonstrated that the post-LOCA CR doses are maximized at the lower inleakage rate due to the additional CR envelope atmosphere cleaning rate offered by higher inleakage rates after the end of the positive pressure period (PPP) primary containment release.

2. Assumes conservative unfiltered inleakage of 1000 cfm (Reference 17). For TSC dose calculations, higher inleakage values are marginally conservative because the larger unfiltered inleakage results in higher doses during the controlling normal clean-up

mode of operation during the first 30 minutes of the event (Reference 16), i.e., prior to isolation.

3. Whole Body, Thyroid, or Skin dose in rem, current regulatory limit in parenthesis.

Table 3-3: MSLBA Radiological Consequence Analysis, Offsite Doses

Case	**EAB[1]** **(rem TEDE)**	**LPZ[2]** **(rem TEDE)**	**Regulatory Limit[4]** **(rem TEDE)**
2 μCi/gm dose equivalent I-131	0.67	0.19	25
0.2 μCi/gm dose equivalent I-131	6.7E-02	1.9E-02	2.5
Current Licensing Basis[3]	0.00058 Whole Body 0.27 Thyroid	Not Calculated	25 Whole Body 300 Thyroid

1. Worst 2-hour integrated dose.
2. 30-day integrated dose.
3. Whole Body and Thyroid dose in rem.
4. Regulatory Limits in accordance with RG 1.183

Table 3-4: MSLBA Radiological Consequence Analysis, CR Doses

Case	CR Dose[1,2] (rem TEDE)	Regulatory Limit (rem TEDE)
2 μCi/gm dose equivalent I-131	2.61	5
0.2 μCi/gm dose equivalent I-131	0.26	5
Current Licensing Basis	Not Calculated	-

1. Assumes a conservative unfiltered inleakage of 0 cfm. For the CR MSLBA calculations, a lower inleakage is conservative because the source is a limited release over 10.5 seconds.

2. Assumes CR Emergency Mode Ventilation rate of 900 cfm (Reference 40).

Table 3-5: FHA Radiological Consequence Analysis

Fuel Handling Accident **CR Inleakage**	**EAB[1]** **(rem TEDE)**	**LPZ[2]** **(rem TEDE)**	**CR Operator** **(30 day)**	**TSC Operator** **(30 Day)**
FHA 1000 CFM	0.94	0.23	3.16	2.83
0 CFM	0.94	0.23	2.31	1.71
Regulatory Limit	6.25	6.25	5.00	5.00
Current Licensing Basis[3]	0.011 (6) Whole Body 0.006 (75) Thyroid	0.036 (6) Whole Body 0.038 (75) Thyroid	Not previously analyzed	Not previously analyzed

1. Worst 2-hour integrated dose.

2. 30-day integrated dose.

3. Whole Body and Thyroid dose in rem, current regulatory limit in parenthesis.

Table 3-6: CRDA Radiological Consequence Analysis, Offsite Doses

Event	EAB[1] (rem TEDE)	LPZ[2] (rem TEDE)
Control Rod Drop Accident	0.06	0.04
Regulatory Limit	6.25	6.25
Current Licensing Basis[3]	0.29 (6) Whole Body 9.65 (75) Thyroid	0.07 (6) Whole Body 5.29 (75) Thyroid

1. Worst 2-hour integrated dose.
2. 30-day integrated dose.
3. Whole Body and Thyroid dose in rem, current regulatory limit in parenthesis.

Table 3-7: NUREG-0737, Item II.B.2 Assessment

Parameter	**Whole Body Dose[1] (rem TEDE)**	**Extremities[1] (rem)**
PASS Sampling Mission Dose	4.66	24.57
Regulatory Limit	5	75
Current Licensing Basis[2]	4.46	22.83

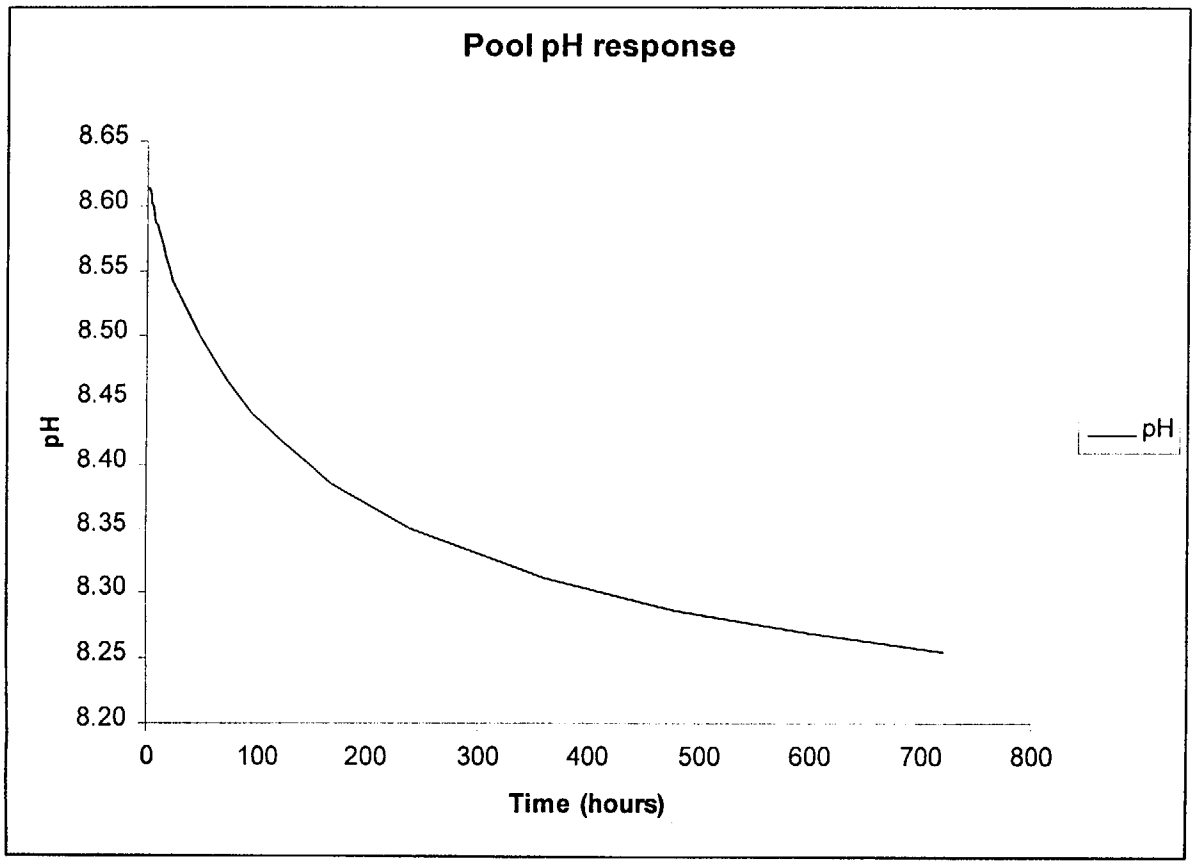
1. Total mission integrated dose.

2. Whole Body REM.

Table 3-8: Acid Generation and Calculated Pool pH Values

Time (Hours)	HI moles	HNO3 moles	HCl moles	pH
2	0.81	9	123	8.61
4	0.82	15	190	8.60
8	0.82	23	294	8.59
16	0.82	32	457	8.56
24	0.82	39	592	8.54
48	0.82	53	882	8.50
72	0.82	62	1098	8.47
120	0.82	74	1413	8.42
168	0.82	84	1634	8.39
240	0.82	96	1868	8.35
480	0.82	123	2284	8.29
720	0.82	139	2484	8.25

Figure 3-1: Pool pH Response



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