VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

April 11, 2000

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 00-123
SPS-LIC/CGL R0
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS AND BASES CHANGE -
ALTERNATE SOURCE TERM IMPLEMENTATION

In a letter dated January 10, 2000 (Serial No. 99-620), Virginia Electric and Power Company restated the intention to participate in the Alternate Source Term (AST) pilot program for Surry Power Station Units 1 and 2. We have completed our reanalysis of the Loss of Coolant Accident and the Fuel Handling Accident using the AST, and this letter provides the license amendment request for AST implementation as the plant design and licensing bases for Surry.

Pursuant to 10CFR50.90, Virginia Electric and Power Company requests amendments, in the form of revisions to the Technical Specifications to Facility Operating License Numbers DPR-32 and DPR-37 for Surry Power Station Units 1 and 2. This change is requested in accordance with the requirements of 10CFR50.67, which addresses the use of an AST at operating reactors. The proposed change to implement the AST revises Technical Specifications (TSs) 3.7, 3.10, and 3.22, as well as the Bases of TSs 3.4, 3.8, 3.10, 3.19, and 3.22. The change includes the following revisions:

- Allows a slight atmospheric pressure (0.5 psig) in containment during the one to four hour interval following a Loss of Coolant Accident with subatmospheric pressure being reached within four hours (TS 3.4 Basis, TS 3.8 Basis, TS 3.19 Basis)

- Deletes the automatic function requirements and setpoints for the containment particulate and gas monitors, as well as the manipulator crane area monitors (TS Table 3.7-5)

- Revises the Applicability and Objective statements to also include irradiated fuel movement in the Fuel Building (TS 3.10, TS 3.10 Basis)

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- Delineates which conditions apply during refueling operations or during irradiated fuel movement in the Fuel Building (TS 3.10, TS 3.10 Basis)

- Revises the requirements for the equipment access hatch, the personnel airlock, and penetrations having a direct path to the outside atmosphere to be capable of being closed (TS 3.10, TS 3.10 Basis)

- Deletes the requirement for testing and operability of the Containment Purge System and automatic isolation of this system during refueling (TS 3.10, TS 3.10 Basis)

- Revises the requirement for operability and continuous monitoring of the manipulator crane area monitors, containment particulate and gas monitors, fuel pit bridge radiation area monitor, and ventilation vent stack 2 particulate and gas monitors for identification of the occurrence of a fuel handling accident (TS 3.10, TS 3.10 Basis)

- Deletes the requirement to filter fuel building exhaust during refueling (TS 3.10, TS 3.10 Basis)

- Deletes the requirement to filter containment purge exhaust during refueling (TS 3.10, TS 3.10 Basis)

- Adds requirements to have two trains of the control room bottled air system operable during refueling operations and during irradiated fuel movement in the Fuel Building, as well as actions for inoperability (TS 3.10); this requirement parallels the existing emergency ventilation system requirement in TS 3.10

- Adds clarification of the requirement to cease refueling operations if the limiting conditions are not met (TS 3.10)

- Deletes the requirement to manually realign auxiliary ventilation from the refueling mode on a safety injection signal (applicable for all fuel handling versus decayed fuel) (TS 3.22, TS 3.22 Basis)

A discussion of the proposed Technical Specifications and Bases change is provided in Attachment 1.

The proposed Technical Specifications and Bases change has been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee. It has been determined that the proposed Technical Specifications and Bases change does not involve an unreviewed safety question, as defined in 10CFR50.59. The proposed Technical Specifications and Bases change mark-up and typed pages are provided in Attachments 2 and 3, respectively. The basis for our determination that the Technical Specifications change does not involve a significant hazards, as defined in 10CFR50.92, is provided in Attachment 4.

As noted in our January 10, 2000 letter, we request that the review fees associated with the NRC evaluation of this license amendment submittal be waived. This request is made pursuant to 10CFR170.11(b)(1), which governs exemptions from fees granted upon the initiative of the NRC. This request is based on 1) the participation of Surry Power Station as a pilot plant and as a member of the NEI Task Force that supported the development of the proposed rule and associated regulatory guide and 2) the technical information and support provided by Virginia Power for the Surry units which were analyzed during the NRC re-baselining analysis effort associated with the AST development work.

Should you have any questions or require additional information, please contact us.

Very truly yours,

David A. Christian
Vice President – Nuclear Operations

Attachments:
1. Discussion of Change
2. Mark-up of Technical Specifications and Bases
3. Proposed Technical Specifications and Bases Change
4. Significant Hazards Consideration Determination

Commitments made in this letter: None.

cc: U.S. Nuclear Regulatory Commission
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by David A. Christian, who is Vice President - Nuclear Operations, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 11th day of April, 2000.

My Commission Expires: 3/31/04.

Maggie McClure
Notary Public

(SEAL)

Attachment 1

Discussion of Change

ASSESSMENT OF ACCIDENT RADIOLOGICAL CONSEQUENCES USING NUREG-1465 METHODOLOGY

SURRY POWER STATION UNITS 1 AND 2

VIRGINIA POWER

March 2000

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1.0 Introduction & Background
1.1 Introduction

This report describes the evaluations conducted to assess the radiological consequences of implementing the NUREG-1465 (1) accident source term methodology for Surry Units 1 and 2. The accident source term documented in Reference (1) is herein referred to as the Alternative Source Term (AST). This convention is adopted following that originated by the NRC staff in the rulemaking proceeding associated with application of AST technology. The NRC, in Reference (2), issued the final rule and draft regulatory guidance associated with use of alternative source terms at operating reactors. The discussion in this report provides justification for the license amendment request, per the provisions of newly issued CFR § 50.67, 'Accident Source Term.' This request for Surry Units 1 and 2 is submitted for consideration as a pilot plant application, in conjunction with the NRC and Nuclear Energy Institute's program for AST implementation. This is consistent with the intention for submitting such an application stated in Reference (20).

The evaluations documented herein have in general employed the detailed methodology proposed in Draft Regulatory Guide DG-1081 (3) for use in design basis accident analyses for alternative source terms. Where alternative approaches to those specified in DG-1081 are proposed, supporting justification is provided for the NRC staff's use in making a determination of the acceptability of such approaches.

Certain aspects of this application, if granted and implemented, will allow increased operational flexibility and efficiency, reduction in regulatory burdens and potential reduction in calculated radiological doses for specific design basis accidents.

1.2 Current Licensing Basis Summary

The current design basis accident radiological assessments that appear in the Surry Power Station Updated Final Safety Analysis Report (UFSAR) were conducted in support of a license amendment to increase the core rated thermal power. These analyses were performed by Virginia Power with the

Bechtel LOCADOSE code (4). The radiological analysis description, which included offsite and control room doses, was submitted to NRC in August 1994 via Reference (5). The NRC staff SER approving the core power increase was issued in August 1995 (6).

The existing design basis accident radiological analyses consist of assessments for the following events, which employ the analytical guidance as cited below:

1) Loss of Coolant Accident (Regulatory Guide 1.4; NUREG-0800, Section 15.6.5)
2) Main Steam Line Break (NUREG-0800, Section 15.1.5)
3) Steam Generator Tube Rupture (NUREG-0800, Section 15.6.3; WOG Methodology (7))
4) Locked Rotor Accident (NUREG-0800, Sections 15.3.3, 15.3.4)
5) Fuel Handling Accident (NUREG-0800, Section 15.7.4; Regulatory Guide 1.25)
6) Waste Gas Decay Tank Rupture (NUREG-0800, Branch Technical Position ETSB 11-5)
7) Volume Control Tank Rupture (NUREG-0800, Branch Technical Position ETSB 11-5)

The existing analyses for these events assume the radiological source term documented in TID-14844 (8) and dose conversion factors that are consistent with those in Regulatory Guide 1.109 (9). Table 1.2-1 provides a summary of results from the first five events above for information. The last two events have minimal dose consequences, with the whole body exposure calculated to be less than 0.5 rem at the EAB.

Table 1.2-1

Summary of Significant Radiological Results
Using TID-14844 Source Term and Current Analysis Methodologies
Surry Power Station Units 1 and 2

Accident	Control Room Dose (rem)		EAB Dose (rem)		LPZ Dose (rem)	
	Thyroid	Whole Body	Thyroid	Whole Body	Thyroid	Whole Body
LOCA	29.0	0.2	224.0	6.0	12.0	0.3
Main Steamline Break	3.6	< 0.1	3.6	< 0.1	0.4	< 0.1
SG Tube Rupture	8.1	< 0.1	15.4	< 0.1	0.7	< 0.1
Locked Rotor	10.6	0.2	2.1	0.3	0.7	< 0.1
Fuel Handling	2.4	0.1	55.0	1.6	2.4	0.1

1.3 Analysis Assumptions & Key Parameter Values

1.3.1 Selection of Events Requiring Reanalysis

A full implementation of the AST (as defined in Section 1.2.1 of Reference 3) is proposed for Surry Units 1 and 2. To support the licensing and plant operation changes discussed in Section 2.0, the Loss of Coolant Accident (LOCA) and Fuel Handling Accident (FHA) were reanalyzed employing the NUREG-1465 source term. The analysis methodology generally applied the guidance of DG-1081, in conjunction with the total effective dose equivalent (TEDE) calculational methodology. If this request is granted, the source term documented in NUREG-1465, as implemented in this plant-specific application, will become the source term employed in design basis radiological analyses for Surry Units 1 and 2.

The proposed licensing and plant operational changes are discussed in Section 2.0. A summary of the key plant operational changes is provided here for use in illustrating the logic used to determine the accident analyses that were impacted. These changes require appropriate changes to the Surry Technical Specifications, which are described in Section 2.0 of this report. The key changes considered in determining which accidents were reanalyzed are listed below:

a. eliminate credit for filtration of effluents from post-accident ECCS leakage

b. eliminate credit for filtration of fuel building and containment exhaust during refueling

c. allow an open equipment access hatch, containment personnel airlock & certain containment penetrations during refueling

d. allow positive containment pressure for up to four hours after DBA (versus current limit of one hour)

e. eliminate the automatic containment purge isolation requirements during refueling

As indicated in Section 1.2.1 of Reference (3), the design basis LOCA must be reanalyzed to support an application for full implementation of the AST. The ECCS filtration (Item a) and positive containment pressure (Item d) changes above would also impact the LOCA accident dose results.

Item a – radioactive leakage from ECCS components only occurs following the transition to recirculation cooling mode in which contaminated water is circulated from the containment sump through portions of the ECCS and Recirculation Spray systems that are outside containment. The proposed change assumes no filtration of the airborne activity from the ECCS component leakage. The design basis LOCA accident is the only Surry event for which radiological consequences are analyzed which is impacted by this change.

Item b – the exhaust from containment and the fuel building is currently filtered during refueling operations that have the potential to cause damage to fuel, either during fuel movements or movement of other components. The proposed change eliminates the requirement for this filtration. This change only impacts the Fuel Handling Accident (FHA). No other events for which radiological consequences are calculated have release paths that are directed through these filtration systems during refueling operations.

Item c – the containment equipment access hatch, at least one door of the personnel airlock and other containment penetrations are currently closed during refueling operations. This ensures that these do not represent release pathways for radioactive material. The proposed change would allow these pathways to be open, but with a requirement to be capable of being closed. The only significant source of radioactive release during refueling is from a Fuel Handling Accident that breaches fuel cladding. This change only impacts the Fuel Handling Accident.

Item d – the current subatmospheric containment design basis requires that the engineered safeguards systems act to depressurize containment to less than atmospheric pressure within one hour and to maintain subatmospheric conditions thereafter. The proposed change would allow the calculation of pressures slightly above atmospheric pressure for a limited duration (1 – 4 hours) after the design basis event. This change could potentially impact either the design basis LOCA or main steamline break events. Since only the LOCA event has significant radiological releases into containment, it is the only analyzed event impacted by this change.

Item e – the current Technical Specifications require that during refueling, the isolation valves and associated radiation monitors in the Containment Ventilation and Purge system be operable to isolate purge flow pathways on a high radiation condition. The proposed changes eliminate the requirement for operability of the automatic purge isolation function. The associated radiation monitors will still be relied upon for identification of a Fuel Handling Accident. This change, which involves the potential open pathways in containment, only affects the analysis of the Fuel Handling Accident inside containment.

It can be concluded from this evaluation summarized above that for implementing the AST in conjunction with the proposed plant operational changes, only the LOCA and Fuel Handling Accidents require reanalysis. Sections 3.1 and 3.2, respectively, provide the detailed description of the reanalyses for these events. Section 3.3 documents an evaluation of the radiological analyses for the remaining events which supports the conclusion that results of the unanalyzed events remain acceptable for implementation of the AST.

1.3.2 Analysis Assumptions & Key Parameter Values

This section describes the general analysis approach and presents analysis assumptions and key parameter values that are common to the accident analyses performed to implement the NUREG-1465 source term. Sections 3.1 and 3.2 provide specific assumptions that were employed for the LOCA and FHA analyses, respectively.

The dose analyses documented in this application employ the Total Effective Dose Equivalent (TEDE) calculational method, consistent with the radiation protection standards in 10 CFR Part 20 and as specified in DG-1081 for AST applications. The TEDE concept is defined to be the deep dose equivalent, DDE, (from external exposure) plus the committed effective dose equivalent, CEDE, (from internal exposure). In this manner, the TEDE dose assesses the impact of all relevant nuclides upon all body organs, in contrast with the previous single, critical organ (thyroid) concept for assessing internal exposure.

The definition of source term, as presented in CFR § 50.67, states '*source term* refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.' Footnote 1 to CFR § 50.67(b)(1) clarifies that the source term to be assumed in radiological consequence analyses of design basis accidents '... should be based upon a major accident, hypothesized for the purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.'

These statements clearly are most applicable to the source term employed for analyses of the design basis LOCA event. The AST characteristics assumed in the Surry LOCA analysis documented herein conform to these requirements. It is not as apparent, however, what these definitions imply regarding the applicable assumptions for implementing the AST analysis of less severe accidents. One principle that can be derived from the statement in the footnote is that the predicted consequences for a given design basis accident radiological analysis not be underpredicted for any event sequence that is considered credible. This is not the same as stating that the analysis assumptions should define a sufficiently <u>incredible</u> sequence, from which extremely conservative radiological consequences would be obtained. This principle of conservatively bounding event sequences that are considered credible has been applied in the Surry reanalyses.

There are a number of analysis assumptions and plant features that are used in the analysis of both the LOCA and FHA events. These items are presented in Table 1.3-1.

Table 1.3-1

Analysis Assumptions & Key Parameter Values Employed in Both LOCA and FHA Analyses

<u>NSSS Parameters</u>	
Core Power	2605 MWt
Number of Fuel Assemblies	157
Containment Free Volume	1.863E6 ft^3
<u>Main Control Room (MCR) Parameters</u>	
Free Volume	2.23E5 ft^3
Emergency Ventilation Intake Flow	1000 cfm
Emergency Ventilation Recirculation Flow	0 cfm
Emergency Ventilation Air Bottles-Actuation Time	0 seconds[1]
Emergency Ventilation Intake-Actuation Time	60 minutes
Unfiltered Inleakage	10 cfm
Emergency Ventilation Intake Filtration Efficiency	
Elemental Iodine	90%
Organic Iodine	70%
Particulate (aerosol) Iodine	99%
<u>Offsite Atmospheric Dispersion Factors</u>	
Exclusion Area Boundary, EAB (0 – 2 hours)	3.40E-3 sec/m^3
Exclusion Area Boundary, EAB (2 – 8 hours)	1.85E-3 sec/m^3
Low Population Zone, LPZ	
0 – 8 hours	1.66E-4 sec/m^3
8 – 24 hours	9.76E-5 sec/m^3
24 – 96 hours	3.06E-5 sec/m^3
96 – 720 hours	5.79E-6 sec/m^3
<u>Breathing Rates</u>	
Control Room	3.47E-4 m^3/sec
Offsite (EAB & LPZ)	
0 – 8 hours	3.47E-4 m^3/sec
8 – 24 hours	1.75E-4 m^3/sec
24 – 720 hours	2.32E-4 m^3/sec

[1] System is effective from the start of the accident, actuated on either an SI signal (LOCA) or manual actuation upon detection of fuel handling accident (FHA).

Table 1.3-1 (continued)

Analysis Assumptions & Key Parameter Values Employed in Both LOCA and FHA Analyses

Control Room Occupancy Factors	
0 – 24 hours	1.0
24 – 96 hours	0.6
96 – 720 hours	0.4

Key Operator Actions

Initiate 1 Fan of Main Control Room Emergency Ventilation Intake at 60+ Minutes After Accident

2.0 Proposed Licensing Basis Changes

This section provides a summary description of the key proposed licensing basis changes that are justified with the Surry AST analyses accompanying this license amendment request.

2.1 Implementation of NUREG-1465 Methodology as Design Basis Source Term

This report supports a request to revise the design basis accident source term for Surry Units 1 and 2. Subsequent to approval of this license amendment, the design basis source term for use in evaluating the consequences of design basis accidents will become the source term documented in NUREG-1465 (1), including any deviations approved by the NRC staff. This license amendment application is made pursuant to the requirements of CFR § 50.67(b)(1), which specifies that any licensee seeking to revise its current accident source term used in design basis radiological consequences analysis shall apply for a license amendment.

2.2 Open Personnel Air Lock, Equipment Access Hatch & Penetrations During Refueling

This change is an example of a cost reduction and operational enhancement that is made possible by implementing the AST for Surry. Currently, Technical Specifications 3.10, Refueling, requires that the equipment access hatch and at least one door in the personnel airlock be closed during refueling operations. In addition, penetrations that provide a direct path from containment atmosphere to the outside atmosphere must have operable isolation valves or be closed. This requirement is consistent with the existing analysis for a fuel handling accident inside containment, which does not model radioactive releases through these pathways. The existing requirements, however, hinder efficient movement of personnel in and out of containment during refueling operations, involve cycling of the personnel airlock doors for each containment entry and require other involved activities to manage containment penetrations. This leads to increased wear and maintenance on the airlock and inefficiency of operations.

The proposed change will increase the efficiency of operations and reduce wear upon the airlock mechanisms. Because there could be a large number of personnel in containment during refueling operations, it may take several cycles of the airlock to evacuate all personnel in the event of a fuel handling accident. This additional time required for evacuation would increase personnel doses. The proposed Technical Specifications changes require that the equipment access hatch, at least one door in the personnel airlock and any open containment penetrations be capable of being closed. The penetrations that are allowed to be open are those that terminate in the Auxiliary Building or Safeguards and provide a direct path between containment atmosphere and outside atmosphere. Changes to operating procedures will be implemented to ensure that the capability to close these openings is maintained during refueling operations and that the required actions can be accomplished.

Closure of the equipment access hatch is the duty of a team trained for that task and controlled in accordance with station procedures. Equipment hatch closure will be accomplished as allowed by containment dose rates, and may require containment entry after the personnel airlock has been closed. Since the revised radiological analysis does not take credit for the containment closure actions, no commitment is being proposed concerning the required timeframe for achieving containment closure. This represents an exception to the guidance proposed in DG-1081, which recommends an assumed 30 minute closure time. Furthermore, in the case of the equipment access hatch, it could potentially pose an unacceptable personnel radiological hazard if prompt closure was required following a fuel handling accident inside containment. To preclude creating such a hazard, closure will only be accomplished as allowed by containment dose rates.

2.3 Eliminate Filtration of Containment & Fuel Building Exhaust During Refueling

This change is another example of operational efficiency that is achievable from implementing the AST analyses. Currently, Technical Specifications 3.10, Refueling and the basis for 3.22, Auxiliary Ventilation Exhaust Filter Trains, require that the fuel building exhaust and the containment purge exhaust be continuously filtered through safety-related high efficiency particulate air (HEPA) filters and charcoal adsorbers during refueling operations. This requirement is consistent with the existing analysis for a fuel handling accident inside containment, which assumes reduced radiological releases associated with this filtration. The revised radiological analyses of the Fuel Handling Accident take no credit for operation of the HEPA filters or charcoal adsorbers to reduce the radioactive content of releases from either containment or the Fuel Building. The AST license amendment proposes changes to Technical Specification 3.10 and 3.22 that remove the requirement to filter both containment purge and fuel building exhaust through these filters.

2.4 Redefinition of Subatmospheric Containment Depressurization Criteria

This change proposes a relaxation of the current containment design basis acceptance criteria concerning achieving and maintaining subatmospheric conditions following a loss of coolant accident. Surry Units 1 and 2 have a subatmospheric containment design, that has the following acceptance criteria for the design basis LOCA containment integrity analyses:

- calculated peak pressure must be less than 45 psig
- containment must be depressurized to less than atmospheric within 1 hour
- calculated peak pressure after one hour must be less than 0.0 psig

The second and third criteria are being relaxed as part of the present application. The proposed acceptance criteria for design basis LOCA containment integrity analyses are as follows (the first item remains unchanged):

- calculated peak pressure must be less than 45 psig
- containment must be depressurized to 0.5 psig within 1 hour and to subatmospheric pressure within 4 hours
- calculated peak pressure after 4 hours must be less than 0.0 psig

The current criteria require that following the initial containment depressurization to less than atmospheric pressure, operation of the Recirculation Spray subsystems indefinitely maintains pressure less than atmospheric. These criteria are currently reflected in the Bases of the following Surry Technical Specifications: TS 3.4, Spray Systems; TS 3.8.D, Containment-Internal Pressure;

TS 3.19, Main Control Room Bottled Air System. The AST license amendment proposes changes to the bases for each of these three Technical Specifications to indicate the relaxed pressure criterion at 1 hour and the extension of the requirement to achieve subatmospheric pressure until 4 hours. The radiological analyses have accommodated greater than atmospheric pressure and the associated period of additional leakage for an interval of up to 4 hours after the DBA. The analyses for implementation of the AST for Surry have assumed a containment leakage rate that corresponds to a maximum containment pressure of 0.5 psig for the timeframe of 1 to 4 hours following the loss of coolant accident and zero leakage thereafter. Section 3.1 provides the detailed justification for the leakrate assumed in the analysis of the LOCA.

The change in the subatmospheric design basis was reviewed to confirm that no additional design basis considerations (beyond radiological effects of the change) were impacted. This review has concluded that no additional considerations are involved that are not assessed by including the increased containment leakage in the radiological analysis. This is consistent with the original licensing evaluation of the Surry subatmospheric design concept, as documented by the NRC in the Surry SER (22). Section 3.2.2.3, 'Containment Subatmospheric Concept' of the SER states:

> *"We have analyzed the consequences of the loss-of-coolant accident presented in Section 3.1.9.2 of this evaluation assuming the containment leaks at its design leakage rate of 0.1% per day for a period of 60 minutes following a loss-of-coolant accident, and that no out-leakage occurs thereafter. Based on our evaluation of the analytical techniques used by the applicant to calculate depressurization time, we have concluded that the increase in depressurization time from the 38 minutes calculated by the applicant to the 60 minutes used in the staff analysis represents a conservative estimate of the maximum length of time out-leakage could occur."*

There are no proposed changes to the existing containment structure, heat removal systems, containment integrity accident analyses or Technical Specifications associated with these items as part of this application. The proposed changes are intended to provide potential future flexibility by utilizing a portion of the margin that was made available by application of the AST analysis methodology.

2.5 Eliminate Containment Purge Isolation Operability Requirement for Refueling

This change involves eliminating the requirement to maintain an operable automatic isolation capability for the Containment Ventilation Purge system during refueling. Automatic isolation occurs in response to high radiation signals from containment area and airborne radiation monitors. Currently, Technical Specifications 3.10, Refueling, requires testing of this system and the associated radiation monitors immediately prior to refueling. The existing analysis for a fuel handling accident inside containment assumes failure of the purge isolation function and therefore already models radioactive releases through the purge pathway. The AST analysis methodology allows releases through this pathway, but with calculated doses that are less than in existing analyses. This change is proposed to provide flexibility in refueling operations. The revised radiological analyses of the Fuel Handling Accident take no credit for operation of the purge isolation function by accommodating continued forced ventilation flow through this pathway. The AST license amendment proposes changes to Technical Specification 3.10 to eliminate the requirement for operability of the purge isolation function, while retaining operability requirements for the radiation monitors (to provide fuel handling accident identification). It is proposed that the radiation monitor setpoints presently in Technical Specification Table 3.7-5 be relocated to another licensee controlled document.

3.0 Radiological Event Reanalyses & Evaluation

As documented in Section 1.3.1, this application involves the reanalysis of the design basis radiological analyses for the LOCA and Fuel Handling Accidents (FHA). These analyses have incorporated the features of the AST, including the TEDE analysis methodology and modeling of plant systems and equipment operation that influence the events. The calculated radiological consequences are compared with the revised limits provided in 10 CFR 50.67(b)(2), as clarified per the additional guidance in DG-1081 for the FHA event. Dose calculations are performed for the exclusion area boundary (EAB) for the worst 2 hour period, and for the low population zone (LPZ) and control room for the duration of the accident (30 days). All the radiological consequence calculations for the AST were performed by Virginia Power with the LOCADOSE computer code system (4). The LOCADOSE codes were developed by Bechtel Corporation to analyze doses from transport of radioactive materials through multi-region systems. The dose acceptance criteria that apply for implementing the AST are provided in Table 3.1-1.

Table 3.1-1 – Accident Dose Acceptance Criteria

Accident or Case	Control Room	EAB & LPZ
Design Basis LOCA	5 rem TEDE	25 rem TEDE
Steam Generator Tube Rupture		
Fuel Damage or Pre-incident Spike	5 rem TEDE	25 rem TEDE
Coincident Iodine Spike	5 rem TEDE	2.5 rem TEDE
Main Steam Line Break		
Fuel Damage or Pre-incident Spike	5 rem TEDE	25 rem TEDE
Coincident Iodine Spike	5 rem TEDE	2.5 rem TEDE
Locked Rotor Accident	5 rem TEDE	2.5 rem TEDE
Rod Ejection Accident	5 rem TEDE	6.25 rem TEDE
Fuel Handling Accident	5 rem TEDE	6.25 rem TEDE

3.1 Large Break Loss of Coolant Accident (LOCA) Reanalysis

This section describes the methods employed in and results obtained from the LOCA design basis radiological analysis. The analysis includes dose from several sources: the containment leakage plume and leakage from ECCS components that persists throughout the assumed 30 day duration of the accident. Doses were calculated at the exclusion area boundary (EAB), at the low population zone boundary (LPZ), and in the control room. The methodology used to evaluate the control room and offsite doses resulting from a LOCA was consistent with DG-1081 (3).

3.1.1 LOCA Scenario Description

The design basis LOCA scenario for radiological calculations is initiated assuming a major rupture of the primary reactor coolant system piping. In order to result in radioactive releases of the magnitude specified in NUREG-1465, it is also assumed that the emergency core cooling system does not provide adequate core cooling, such that significant core melting occurs. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the NUREG-1465 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design basis transient analysis.

3.1.2 LOCA Source Term Definition

NUREG-1465 (1) provides explicit description of the key AST characteristics recommended for use in design basis radiological analyses. There are significant differences between the source term in Reference (1) and the existing design basis source term documented in TID-14844 (8). The primary differences between the key characteristics of the two source terms are shown in Table 3.1-2 below.

Table 3.1-2 – Comparison of TID-14844 and NUREG-1465 Source Terms

Characteristic	TID Source Term	NUREG-1465 Source Term
Core Fractions Released To Containment	Noble gases – 100% Iodine – 50% (half of this plates out) Solids – 1% Iodine – 50% to sump	Noble gases – 100% Iodine – 40% Cesium – 30% Tellurium – 5% Barium – 2% Others – 0.02% to 0.2%
Timing of Release	Instantaneous	Released in Two Phases Over 1.8 hour Interval
Iodine Chemical and Physical Form	91 % inorganic vapor 4% organic vapor 5% aerosol	4.85% inorganic vapor 0.15% organic vapor 95% aerosol
Solids	Ignored in analysis	Treated as an aerosol

NUREG-1465 divides the releases from the core into two phases: 1) the fuel gap release phase during the first 30 minutes and 2) the early in-vessel release phase in the subsequent 1.3 hours. The later release phases documented in NUREG-1465 are not considered for design basis accidents, consistent with the guidance from DG-1081. Table 3.1-3 shows the fractions of the total core inventory of various isotope groups assumed to be released in each of the two phases of the LOCA analysis. Table 3.1-3 also shows the rate of release or production for each isotope group, assuming that the releases are linear with respect to time.

Table 3.1-3 – NUREG-1465 Release Phases

	Core Release Fractions		Production Rate (Frac/hr)[a]	
Isotope Group	Gap	Early In-Vessel	Gap	Early In-Vessel
Noble Gases[b]	0.05	0.95	0.1	7.31E-01
Halogens	0.05	0.35	0.1	2.69E-01
Alkali Metals	0.05	0.25	0.1	1.92E-01
Tellurium	0	0.05	0	3.85E-02
Barium, Strontium	0	0.02	0	1.54E-02
Noble Metals	0	0.0025	0	1.92E-03
Cerium	0	0.0005	0	3.85E-04
Lanthanides	0	0.0002	0	1.54E-04
Duration (hr)[a]	0.5	1.3		

a. Release duration and production rates apply only to the Containment release. The ECCS leakage portion of the analysis conservatively assumes that the entire core release fraction is in the containment sump from the start of the LOCA.

b. Noble Gases are not scrubbed from the containment atmosphere and therefore are not found in either the sump or ECCS fluid.

The core radionuclide inventory for use in determining source term releases was generated using the ORIGEN2 code. The calculations are based on representative design characteristics for the low-leakage cores operated in the Surry units and an assumed power level of 2605 MWt. This assumed power slightly exceeds 102% of the licensed core rated thermal power of 2546 MWt. Table 3.1-4 lists the isotopes and the associated total core activities at the end of a fuel cycle. Also shown in Table 3.1-4 are the inhalation and immersion dose conversion factors for each of the isotopes. These dose conversion factors are for use in determining the dose in TEDE units and are taken from Reference (10) and (11). The inhalation dose is equivalent to the CEDE dose and the immersion dose is equivalent to the DDE dose discussed in the section of Reference (2) entitled 'I. Background.'

Table 3.1-4 – Core Inventory and Dose Conversion Factors by Isotope

Isotope	Inventory (Ci)	TEDE Dose Conversion Factor	
		Inhalation (Rem/Ci)	Immersion (Rem-m^3/Ci-sec)
I-130	2.45E+06	2.64E+03	3.85E-01
I-131	6.64E+07	3.29E+04	6.73E-02
I-132	9.54E+07	3.81E+02	4.14E-01
I-133	1.35E+08	5.85E+03	1.09E-01
I-134	1.48E+08	1.31E+02	4.81E-01
I-135	1.26E+08	1.23E+03	2.95E-01
I-136	6.01E+07	0.00E+00	0.00E+00
I-137	5.87E+07	0.00E+00	0.00E+00
I-138	2.90E+07	0.00E+00	0.00E+00
Kr-85	8.01E+05	0.00E+00	4.40E-04
Kr-87	3.27E+07	0.00E+00	1.52E-01
Kr-88	4.61E+07	0.00E+00	3.77E-01
Kr-89	5.61E+07	0.00E+00	0.00E+00
Kr-83m	8.12E+06	0.00E+00	5.55E-06
Kr-85m	1.71E+07	0.00E+00	2.77E-02
Xe-133	1.35E+08	0.00E+00	5.77E-03
Xe-135	3.31E+07	0.00E+00	4.40E-02
Xe-137	1.18E+08	0.00E+00	0.00E+00
Xe-138	1.11E+08	0.00E+00	2.13E-01
Xe-131m	7.39E+05	0.00E+00	1.44E-03
Xe-133m	4.21E+06	0.00E+00	5.07E-03
Xe-135m	2.65E+07	0.00E+00	7.55E-02
Cs-134	1.38E+07	4.63E+04	2.80E-01
Cs-136	3.17E+06	7.33E+03	3.92E-01
Cs-137	8.88E+06	3.19E+04	2.86E-05
Cs-138	1.23E+08	1.01E+02	4.48E-01
Cs-139	1.17E+08	0.00E+00	0.00E+00
Cs-134m	3.44E+06	4.37E+01	3.35E-03
Rb-86	1.38E+05	6.62E+03	1.78E-02
Rb-88	4.68E+07	8.36E+01	1.24E-01
Rb-89	6.00E+07	4.29E+01	3.92E-01
Rb-90	5.82E+07	0.00E+00	0.00E+00
Sb-124	9.34E+04	2.52E+04	3.39E-01
Sb-125	1.47E+03	1.22E+04	7.47E-02
Sb-126	3.97E+01	1.17E+04	5.07E-01
Sb-127	7.13E+06	6.03E+03	1.23E-01

Table 3.1-4 – Core Inventory and Dose Conversion Factors by Isotope

Isotope	Inventory (Ci)	TEDE Dose Conversion Factor	
		Inhalation (Rem/Ci)	Immersion (Rem-m^3/Ci-sec)
Sb-129	2.13E+07	6.44E+02	2.64E-01
Te-127	7.09E+06	3.18E+02	8.95E-04
Te-129	2.10E+07	8.95E+01	1.02E-02
Te-131	5.88E+07	4.77E+02	7.55E-02
Te-132	9.39E+07	9.44E+03	3.81E-02
Te-133	7.94E+07	9.21E+01	1.70E-01
Te-134	1.12E+08	1.27E+02	1.57E-01
Te-125m	3.17E+02	7.29E+03	1.68E-03
Te-127m	9.71E+05	2.15E+04	5.44E-04
Te-129m	3.14E+06	2.39E+04	5.74E-03
Te-131m	9.54E+06	6.40E+03	2.59E-01
Te-133m	4.90E+07	4.33E+02	4.22E-01
Ba-139	1.21E+08	1.72E+02	8.03E-03
Ba-140	1.16E+08	3.74E+03	3.17E-02
Ba-141	1.09E+08	8.07E+01	1.54E-01
Ba-136m	5.22E+05	0.00E+00	0.00E+00
Ba-137m	8.41E+06	0.00E+00	1.07E-01
Sr-89	6.44E+07	4.14E+04	2.86E-04
Sr-90	6.30E+06	1.30E+06	2.79E-05
Sr-91	7.78E+07	1.66E+03	1.28E-01
Sr-92	8.45E+07	8.07E+02	2.51E-01
Sr-93	9.60E+07	0.00E+00	0.00E+00
Sr-94	9.09E+07	0.00E+00	0.00E+00
Sr-95	8.43E+07	0.00E+00	0.00E+00
Co-58	1.09E+04	1.09E+04	1.76E-01
Co-60	7.88E+04	2.19E+05	4.66E-01
Mo-99	5.18E+04	3.96E+03	2.69E-02
Pd-109	2.22E+07	1.10E+03	9.29E-04
Rh-105	6.57E+07	9.55E+02	1.38E-02
Rh-106	4.37E+07	0.00E+00	3.85E-02
Ru-103	1.04E+08	8.95E+03	8.33E-02
Rh-103m	9.32E+07	5.11E+00	3.26E-05
Ru-105	7.13E+07	4.55E+02	1.41E-01
Ru-106	3.98E+07	4.77E+05	0.00E+00
Tc-101	1.30E+04	1.79E+01	5.96E-02
Tc-99m	1.06E+08	3.26E+01	2.18E-02
Ce-141	1.09E+08	8.95E+03	1.27E-02

Table 3.1-4 – Core Inventory and Dose Conversion Factors by Isotope

Isotope	Inventory (Ci)	TEDE Dose Conversion Factor	
		Inhalation (Rem/Ci)	Immersion (Rem-m^3/Ci-sec)
Ce-143	1.02E+08	3.39E+03	4.77E-02
Ce-144	9.10E+07	3.74E+05	3.16E-03
Eu-154	5.69E+02	2.86E+05	2.27E-01
Eu-155	3.68E+02	4.14E+04	9.21E-03
Eu-156	8.29E+03	1.41E+04	2.50E-01
La-140	1.19E+08	4.85E+03	4.33E-01
La-141	1.10E+08	5.81E+02	8.84E-03
La-142	1.06E+08	2.53E+02	5.33E-01
La-143	1.01E+08	5.99E+01	1.92E-02
Nb-95	1.15E+08	5.81E+03	1.38E-01
Nb-97	1.12E+08	8.29E+01	1.18E-01
Nb-95m	8.08E+05	2.44E+03	1.08E-02
Nd-147	4.39E+07	6.85E+03	2.29E-02
Pm-147	9.65E+06	3.92E+04	2.56E-06
Pm-148	1.85E+07	1.09E+04	1.07E-01
Pm-149	3.03E+07	2.93E+03	2.00E-03
Pm-151	1.32E+07	1.75E+03	5.59E-02
Pm-148m	2.16E+06	2.26E+04	3.58E-01
Pr-143	1.01E+08	8.10E+03	7.77E-05
Pr-144	9.17E+07	4.33E+01	7.22E-03
Pr-144m	1.09E+06	0.00E+00	1.03E-03
Sm-153	1.95E+03	1.96E+03	8.44E-03
Y-90	6.55E+06	8.44E+03	7.03E-04
Y-91	8.38E+07	4.88E+04	9.62E-04
Y-92	8.48E+07	7.81E+02	4.81E-02
Y-93	9.84E+07	2.15E+03	1.78E-02
Y-94	9.95E+07	6.99E+01	2.08E-01
Y-95	1.07E+08	3.77E+01	1.77E-01
Y-91m	4.52E+07	3.63E+01	9.44E-02
Zr-95	1.15E+08	2.36E+04	1.33E-01
Zr-97	1.11E+08	4.33E+03	3.34E-02
Br-82	3.59E+05	1.53E+03	4.81E-01
Br-83	8.10E+06	8.92E+01	1.41E-03
Br-84	1.40E+07	9.66E+01	3.48E-01
Br-85	1.69E+07	0.00E+00	0.00E+00
Br-87	2.76E+07	0.00E+00	0.00E+00
Br-88	2.94E+07	0.00E+00	0.00E+00

Table 3.1-4 – Core Inventory and Dose Conversion Factors by Isotope

Isotope	Inventory (Ci)	TEDE Dose Conversion Factor	
		Inhalation (Rem/Ci)	Immersion (Rem-m^3/Ci-sec)
Am-241	1.51E+04	4.44E+08	3.03E-03
Am-242	6.27E+06	5.85E+04	2.28E-03
Cm-242	3.47E+06	1.73E+07	2.11E-05
Cm-244	3.22E+05	2.48E+08	1.82E-05
Np-238	2.55E+07	3.70E+04	1.01E-01
Np-239	1.29E+09	2.51E+03	2.85E-02
Pu-238	2.64E+05	3.92E+08	1.81E-05
Pu-239	2.33E+04	4.29E+08	1.57E-05
Pu-240	2.64E+04	4.29E+08	1.76E-05
Pu-241	1.19E+07	8.25E+06	2.68E-07
Pu-243	2.60E+07	1.64E+02	3.81E-03

3.1.3 Determination of Atmospheric Dispersion Factors (X/Q)

3.1.3.1 Onsite (Main Control Room) X/Q

The onsite atmospheric dispersion factors were calculated by Bechtel Power Corporation. Site meteorological data taken over the years 1982-1986 were used in the calculations. The source points modeled were the Unit 1 Containment building and Ventilation Vent No. 2. The ventilation vent is modeled since this is the discharge point for exhaust from the safeguards building and auxiliary building. The receptor points modeled were the turbine building fresh air louvers, the turbine building fresh air intakes and the turbine building rollup doors. These locations represent the potential points for control room air intake.

For onsite receptors, the atmospheric dispersion factors were calculated with the ARCON96 model documented in NUREG/CR-6331 (12). Wake effects were considered in calculating the atmospheric dispersion factors for all the onsite receptor points. It was conservatively assumed that only the portion of the reactor containment dome that is higher than the auxiliary building roof be accounted for in determining the magnitude of the wake effects. Additionally, further conservatism was introduced by only considering one containment dome for wake effect impacts. All releases were modeled as ground-level releases even when the source point was elevated (e.g., Ventilation Vent No. 2). The calculated onsite X/Q values used in the LOCA control room dose analyses are presented in Table 3.1-7.

3.1.3.2 Offsite (EAB & LPZ) X/Q

The offsite atmospheric dispersion factors were also calculated by Bechtel Power Corporation, using the site meteorological data taken over the years 1994-1998. The source point modeled was the Unit 1 Containment building. The receptor points modeled were the Exclusion Area Boundary and Low Population Zone.

The PAVAN model documented in NUREG/CR-2858 (13) was used to calculate the atmospheric dispersion factors for offsite receptors. The "wake-credit not allowed" scenario of the PAVAN results was used, since the closest point of both the EAB and LPZ from the onsite release points is greater than 10 'building heights' of the containment dome (the tallest wake-producing structure). The calculated offsite X/Q values used in the control room dose analyses are presented in Table 1.3-1.

3.1.4 Determination of Containment Spray Iodine Removal Coefficients

There are seven different spray headers belonging to two different systems inside the Surry containment. The Containment Spray system has two separate pump trains. Each Containment Spray pump train supplies a separate circular dome header at the top of containment and a common circular header at the top of the crane wall. The Recirculation Spray System consists of two Inside Recirculation Spray pump trains with one semi-circular header each at the top of the crane wall and two Outside Recirculation Spray pump trains with one semi-circular header each at the top of the crane wall. It is conservative for the analysis of spray removal during LOCA to assume a single failure of one train of engineered safeguards equipment, so that the following analysis assumes one Containment Spray train and one train each of the Inside and Outside Recirculation Spray subsystems are operating.

The containment spray removal rates for aerosol fission products are calculated using the methodology of NUREG/CR-5966 (14), which presents removal equations at 10, 50, and 90 percentile levels. In accordance with guidance in DG-1081, only the 10 percentile (most conservative) equations are used. No credit is taken for iodine plateout.

The removal rates were calculated separately as a function of time for the each of the spray subsystem headers and combined to yield the following effective removal rates for all the sprays:

Table 3.1-5 – Combined Containment and Recirculation Spray Aerosol Iodine Removal Coefficients (λ_{mf})

Aerosol Removal Constant		
Time (hr)		λ_{mf}
From	To	(hr^{-1})
2.78E-02	6.00E-02	3.40E+00
6.00E-02	1.15E-01	7.92E+00
1.15E-01	1.94E-01	1.25E+01
1.94E-01	1.14E+00	1.28E+01
1.14E+00	1.80E+00	9.47E+00
1.80E+00	1.90E+00	6.04E+00
1.90E+00	2.02E+00	4.22E+00
2.02E+00	2.51E+00	2.25E+00
2.51E+00	4.38E+00	1.23E+00
4.38E+00	6.48E+00	1.10E+00
6.48E+00	8.61E+00	1.08E+00
8.61E+00	7.20E+02	1.08E+00

The removal of elemental iodine by sprays continues at a rate of 10 hr^{-1} until a decontamination factor (DF) of 200 is reached, as specified in Section 6.5.2 of NUREG-0800 (15). This DF is reached when the elemental iodine activity in the containment at the end of the early in-vessel release phase is reduced by a factor of 200. The time it takes to achieve this reduction in activity is determined as follows:

$$A = A_0 e^{-\lambda t}$$

$$DF = A_0/A = e^{\lambda t}$$

$$t = \ln(DF)/\lambda = \ln(200)/10 = 0.53 \text{ hr}$$

This is the duration required starting at the end of early in-vessel phase at 1.8 hr. Hence, the post accident time at which elemental iodine removal stops is 2.33 hours (1.80 hr + 0.53 hr). Spray removal of organic iodine is not modeled.

3.1.5 LOCA Analysis Assumptions & Key Parameter Values

3.1.5.1 Special Modeling Considerations

Considerations of margin allocation and Surry system features warranted special modeling attention in certain specific areas. This provided a more appropriate representation of physical phenomena for use in the Surry LOCA radiological analysis. Three such items are discussed in this section: 1) model of containment leakage as a function of containment pressure, 2) model of ECCS backleakage to the Refueling Water Storage Tank (RWST) and 3) auxiliary ventilation system model.

Containment Leakage Model

The following acceptance criteria, replicated from the Section 2.4 discussion, are proposed for this application in modeling the Surry subatmospheric containment design:

- calculated peak pressure must be less than 45 psig
- containment must be depressurized to 0.5 psig within 1 hour and to subatmospheric pressure within 4 hours
- calculated peak pressure after 4 hours must be less than 0.0 psig

The LOCA analysis for implementation of the AST has been performed to conform with these revised acceptance criteria. The LOCA analysis has assumed continued leakage during the 1-4 hour interval after the DBA, but at a diminished rate corresponding to a containment pressure of 0.5 psig. Beyond 4 hours, the pressure is assumed to be less than 0.0 psig, terminating leakage from containment. This section describes the model details for determination of the appropriate leakrate associated with a pressure that is slightly above atmospheric.

To determine the leakage flow from containment as a function of containment pressure, the configuration was modeled as compressible flow through an orifice, sized to allow a flow equal to the design leak rate of 0.1% of volume per day at a pressure of 45 psig. For this situation, it is desired to obtain conservatively large estimated leak rates for pressures less than the design pressure of 45 psig. This is accomplished by selecting the following conservative key inputs for the model: 1) design leak rate at 45 psig; 2) design containment temperature; 3) orifice configuration versus a diffuse 'area source' for containment release.

For this application, the fundamental flow equation is used to develop a ratio of flow conditions at two different containment pressures. This ratio usage allowed simplification of the basic expression such that the leakage flow from containment becomes

$$q2 = q1 \frac{\sqrt{\Delta P2 / \rho 2}}{\sqrt{\Delta P1 / \rho 1}}$$

where q2 is the volumetric containment leak rate for pressures between 45 psig and 0.1 psig, q1 is the volumetric leak rate at 45 psig (0.1 % of the containment volume every 24 hours), ΔP2 is the selected pressure, ρ2 is the density of the air in the containment at the selected pressure, ΔP1 is 45 psig and ρ1 is the density of the air in the containment at 45 psig.

Inserting the design leakrate of 1.29 cfm for Surry, and assuming the containment free volume of 1.863E6 ft^3 indicated in Table 1.3-1, the expression is then evaluated at various postulated containment pressures to determine the resulting leakrates, in cfm. The results of this evaluation are provided in Table 3.1-6. For the interval between 1 and 4 hours after the LOCA, in which the maximum allowed containment pressure is 0.5 psig, the containment leakage is assumed to be constant at 0.270 cfm. This corresponds to a rate of 0.02% of containment volume per day.

Table 3.1-6 - Containment Leakage as a Function of Containment Pressure

Containment Pressure (psig)	Leakage (cfm)
0	0.000
0.1	0.122
0.2	0.173
0.3	0.211
0.4	0.243
0.5	0.270
0.6	0.295
0.7	0.318
0.8	0.339
0.9	0.358
1	0.376
5	0.751
10	0.948
15	1.059
20	1.131
25	1.183
30	1.221
35	1.251
40	1.274
45	1.294

Model of ECCS Backleakage to RWST

Following a design basis LOCA, valve realignment occurs to switch the suction water source for the ECCS from the Refueling Water Storage Tank (RWST) to the containment sump. This action is taken upon level in the RWST reaching a defined setpoint. In this configuration, check valves in the normal suction line from the RWST provide isolation between this contaminated flowstream and the RWST. The LOCA radiological analysis models 640 cc/min leakage flow through these valves into the RWST and release of iodine into the (nearly empty) RWST. This total is intended to accommodate all of the leakage back into the RWST through all paths. Ten percent of the total iodine contained in this leakage is assumed to evolve into the tank.

The pathway for release is not directly from the tank, which is essentially airtight. The top of the RWST contains a vent pipe that discharges into the Safeguards Building sump. Upon receipt of a safety injection signal following a LOCA, the safeguards building exhaust is automatically realigned through the safety-related filters in the Auxiliary Ventilation system, which draws a vacuum in the Safeguards building sump. This ventilation pathway discharges out of Ventilation Vent No. 2. Following switchover of the ECCS to take suction on the containment sump, it is assumed that flow leaks back into the RWST through the ECCS system check valves. The combined effects of ECCS liquid leakage into the tank and the ventilation system drawing from the tank is bounded by assuming that outside air leaks into the RWST at a rate of 10 cfm and that 10 cfm (containing iodine assumed to evolve within the tank) is displaced through the vent pipe into the Safeguards Building, then discharged through Ventilation Vent No. 2 to the atmosphere. Holdup in the RWST is modeled, based on the free tank volume. No credit is taken for filtration of the RWST releases that pass through the Auxiliary Ventilation system and are discharged out of Ventilation Vent No. 2. Main Control Room emergency ventilation intake filtration is modeled, with the assumed filtration efficiencies indicated on Table 1.3-1.

<u>Auxiliary Ventilation System Model</u>

The LOCA analysis model incorporates certain relevant features of the auxiliary ventilation system. This system includes the ventilation and heating systems for the auxiliary building, fuel building, decontamination building, and safeguards areas adjacent to each of the reactor containments. The auxiliary building is a four-level compartmentalized structure containing the auxiliary nuclear equipment for both units. Equipment handling potentially radioactive fluids is located on the lower three levels, isolated and shielded as required. The upper level is a ventilation equipment room.

Within the auxiliary building, three iodine filter assemblies, two safety-related and one non-safety-related, are provided. Each filter bank consists of roughing, HEPA and charcoal filters.

Two safety-related, high-head fans, sized to draw 36,000 cfm each from emergency core cooling system (ECCS) equipment areas through the safety-related filters, are provided. The auxiliary ventilation system exhaust serving the following components is directed through the safety-related filters following a safety injection signal: charging pumps (in cubicles within the auxiliary building), recirculation spray system and low head safety injection pumps (in the safeguards area). Exhaust to the atmosphere is through a common, continuously monitored ventilation vent (Ventilation Vent no. 2) located on the roof of the auxiliary building.

The safety-related filters are designed to provide for removal of elemental and organic iodine that is assumed to evolve from ECCS leakage following a LOCA. The assumed ECCS leakage following a LOCA is provided on Table 3.1-7. As indicated on the table, the leakage that is modeled includes the backleakage into the RWST described in the previous section.

The LOCA analysis model for AST implementation assumes 0% efficiency for the safety-related filters in removing iodine assumed to evolve from the 9600 cc/hr analyzed ECCS leakage. The analysis does credit the general function of the auxiliary ventilation system for providing ventilation and filtration of the air in the vicinity of the charging pump cubicle and Safeguards in order to maintain the current licensing basis of not including the leakage from a passive failure (e.g., pump seal). This degree of dependence upon the filtration is the rationale for not proposing the deletion of Technical Specifications LCOs for operability of the auxiliary ventilation safety-related filters. This is consistent with the current licensing basis for Surry 1 and 2, in which the most credible location for such a failure is postulated as a charging pump, low head SI pump or outside recirculation spray pump seal. Per the guidance stated in Appendix A, Section 5.3 of DG-1081 (3), the auxiliary ventilation system provides a 'ventilation filtration system that exhausts the areas of potential leakage' from a postulated passive failure.

There are a number of additional assumptions and key input parameter values assumed in the analysis of the LOCA cases. Table 3.1-7 presents the most significant of these that are unique to the LOCA analysis for AST implementation.

Table 3.1-7

Analysis Assumptions & Key Parameter Values Employed Only in LOCA Analysis

Containment Parameters	
Cross-Sectional Area	1.25E4 ft^2
Sprayed Volume (60% of total)	1.118E6 ft^3
Unsprayed Volume (40% of total)	7.452E5 ft^3
Mixing Rate – Sprayed to Unsprayed Volume	2 Unsprayed Vol/hr
Sump Volume	5.83E4 ft^3
Containment Leakrate (0 to 1 hour)	0.1% vol per day
Containment Leakrate (1 – 4 hours)	0.021% vol per day
Containment Leakrate (4 hours – 30 days)	0.0% vol per day
ECCS Leakage Parameters	
Fraction of Total Core Iodine Inventory in Sump	0.40
Iodine Transport Time to Sump	Instantaneous
ECCS Leakage Rate (415 sec – 2300 sec)	1928 cc/hr
ECCS Leakage Rate (2300 sec – 30 days)	9600 cc/hr
Iodine Release Fraction (of total in ECCS)	0.10
Physical Form of Released Iodine	97% elemental 3% organic
Auxiliary Building Filtration Efficiency for Released Iodine	0%
Backleakage Rate to RWST via ECCS valves (2300 sec – 30 days)	640 cc/min
Effluent Flowrate from RWST to Atmosphere	10 cfm
RWST Free Volume	53,350 ft^3

MCR Atmospheric Dispersion Factors	Containment	ECCS Leakage
0 – 8 hours	1.69E-3 sec/m^3	1.60E-3 sec/m^3
8 – 24 hours	7.19E-4 sec/m^3	6.99E-4 sec/m^3
24 – 96 hours	1.66E-4 sec/m^3	1.71E-4 sec/m^3
96 – 720 hours	1.20E-4 sec/m^3	1.22E-4 sec/m^3

Key Operator Actions	Timing of Action
Secure Turbine Building Supply Fans Powered from Non-Safety Offsite Power (alters assumed intake point for MCR ventilation)	≤ 24 hours after DBA

Miscellaneous

Offsite Power is Assumed to be Maintained (continued operation of Turbine Building Supply Fans is limiting for MCR ventilation intake)

3.1.6 LOCA Analysis Results

The results of the LOCA dose analysis are presented in Table 3.1-8. These results report the calculated dose for the worst 2-hour interval (EAB), and for the assumed 30 day duration of the event for the control room and LPZ. Separate results are provided for each of the three release pathways considered: containment, ECCS leakage and ECCS backleakage via the RWST. The total dose indicated is the summation of these three components. The doses are calculated with the TEDE methodology, and are compared with the applicable acceptance criteria specified in 10 CFR 50.67 and DG-1081. As indicated on the table, each of the results meets the dose acceptance criteria.

Table 3.1-8 – LOCA Analysis Results

Release Pathway	Control Room Dose (rem TEDE)	EAB Dose (rem TEDE)	LPZ Dose (rem TEDE)
Containment Leakage	0.16	14.60	0.95
ECCS Leakage	0.83	1.75	0.84
RWST Backleakage	1.04	0.06	1.02
Total Dose	2.03	16.41	2.81
Acceptance Criteria	5.0	25.0	25.0

3.2 Fuel Handling Accident (FHA) Reanalysis

This section describes the methods and results employed in the Fuel Handling Accident (FHA) design basis radiological analysis. The analysis includes doses associated with release of gap activity from a fuel assembly either inside containment or in the Fuel Building. Doses were calculated at the exclusion area boundary (EAB), at the low population zone boundary (LPZ), and in the control room. The methodology used to evaluate the control room and offsite doses resulting from the FHA was generally consistent with DG-1081 (3), although some significant exceptions are proposed, with accompanying justification.

3.2.1 FHA Scenario Description

The design basis scenario for the radiological analysis of the FHA assumes that cladding damage has occurred to all of the fuel rods in one fuel assembly. This scenario is unchanged from the assumption in the existing UFSAR analysis. The rods are assumed to instantaneously release their fission gas contents to the water surrounding the fuel assemblies. No detailed mechanism is postulated for such damage, but original design evaluations documented in the UFSAR have concluded that this assumption provides a conservative bound for radiological evaluations of this accident. The analyses include the evaluation of FHA cases that occur in both containment and the Fuel Building, with appropriate modeling for the influence of the different release pathways and operation of ventilation systems.

3.2.2 FHA Source Term Definition

The source term recommended in NUREG-1465 (1) is primarily focused upon description of the key AST characteristics that define an incident with major core melting and release of significant quantities of radioactive material from the fuel. The detailed description from Reference (1) was employed for the modeling of the LOCA analysis documented in Section 3.1 of this report. Reference (1), in Section 3.6, 'Proposed Accident Source Terms,' provides recommended values for use in FHA analyses. The detailed discussion that follows describes a rationale for use of the Reference (1) values in the context of a framework that recognizes the expected variation that exists for source term releases in non-LOCA accident scenarios.

For any event, the amount of radioactive material that is actually released from the fuel is a function of three elements, each of which should be treated in a manner appropriate for the event under consideration:

- Total Available Isotopic Inventory (for the relevant population of rods)
- Fraction of Available Inventory Existing In a Releasable Form
- Release Mechanism (i.e., cladding breach)

For the Surry FHA analysis, these elements are employed in a consistent manner to define an appropriate, but still conservative, amount of radioactive material that can be released from the fuel. This approach deviates from the traditional method of applying bounding values for all parameters, which effectively characterizes all rods in the failed fuel assembly as if they could simultaneously have: 1) the maximum power level, 2) the maximum fission gas release and 3) the maximum isotopic inventory. The simultaneous existence of these characteristics is inherently not physical, which can be demonstrated with the use of available information concerning core design characteristics. The proposed approach relies upon fundamental core design processes and physical relationships that can be quantified during reload core design calculations. Each of the three key elements listed above are described below as applied in the FHA event analysis. These concepts are generally applicable for analysis of other non-LOCA events, provided that event-specific influences are addressed. It is proposed that this approach be used to quantify the source term in future radiological analyses of other non-LOCA events employing the AST for Surry Power Station.

Total Available Isotopic Inventory (for the relevant population of rods)

In the case of the FHA event, it is necessary to quantify the isotopic inventory for the fuel rods in one assembly. For the Surry FHA analysis, the total available isotopic inventory is limited to gaseous isotopes that are not soluble in water that are present after the assumed 100 hour decay period. Only such isotopes could be released from the fuel rod cladding, become airborne above the water surface and represent a radiological source. Applying these criteria yields the following isotopes for consideration:

I-130, I-131, I-132, I-133, I-135
Kr-83m, Kr-85, Kr-85m, Kr-88
Xe-131m, Xe-133, Xe-133m, Xe-135, Xe-135m

It is next necessary to quantify the activity of each isotope, so that the total inventory is defined. The isotopic inventory was first quantified in the aggregate for each of three core regions, defined by cycle of irradiation for the fuel assemblies in that region (i.e., first cycle, second cycle, third cycle). The isotopic inventory of each region was obtained from the same ORIGEN2 core inventory calculation that was described in Section 3.1.2 for the LOCA analysis. The number of assemblies in each region and their radial power distributions were selected to be representative of core design strategies at Surry Units 1 and 2. The results of this calculation demonstrate that a fuel assembly at the end of its first cycle of irradiation contained an inventory of radiologically significant isotopes (primarily iodine) that maximizes the FHA event dose. The specific fraction of this total inventory that is actually in the fuel rod/cladding gap and thus available for release is addressed below.

Fraction of Available Inventory Existing In a Releasable Form

Draft Regulatory Guide DG-1081 (3) provides the following recommended values for the fraction of core inventory in the fuel-clad gap to be assumed for design basis analysis of Non-LOCA accident events:

I-131	0.12
Kr-85	0.15
Other Noble Gases	0.10
Other Halogens	0.10
Alkali Metals	0.10

Key factors that determine the fraction of available inventory in the gap vary considerably among non-LOCA events. Two significant factors are the amount of fuel pellet heatup and transient fission gas release from the fuel. Because of the variability between non-LOCA events with respect to these characteristics, it is considered inappropriate to specify one set of gap fraction values for all non-LOCA events. For the purpose of assessing radiological effects from non-LOCA events, it is proposed to classify them in accordance with the expected amount of fuel heatup. The following classification scheme is proposed:

Category 1 - Events with no transient fuel heatup
(Fuel Handling, Main Steamline Break, SG Tube Rupture)

Category 2 – Events with moderate transient fuel heatup
(Small Break LOCA, Locked Rotor)

Category 3 – Events with significant transient fuel heatup (Rod Ejection/Drop)

Since the amount of fission gas release prior to the onset of cladding damage is strongly affected by the fuel temperature, it is clear that the assumed fission gas release (specified as a fraction of total rod inventory) should vary between each class of events listed above. It is proposed that different gap fraction values be assumed in the radiological analyses of each of the 3 categories of events listed above. The proposed relationship between the values is presented in Table 3.2-1, where '> Category 1' denotes the assumption of a value greater than that listed in the Category 1 column. The values assumed in the Surry analysis of FHA for each of the listed isotope groups equal those in the column for Category 1. Values for other event types are not proposed as part of this application.

Table 3.2-1

Fraction of Core Inventory in Gap for Non-LOCA Event Radiological Analyses

	Event Classification		
Isotope Group	Category 1 (FHA, MSLB, SGTR)	Category 2 (SBLOCA, Locked Rotor)	Category 3 (Rod Ejection/ Rod Drop)
I-131	0.03	> Category 1	> Category 2
Noble Gases[1]	0.03	> Category 1	> Category 2
Other Halogens	0.03	> Category 1	> Category 2
Alkali Metals	0.03	> Category 1	> Category 2

[1] Except Kr-85 value, taken from Reference (21)

The gap fractions in DG-1081 are based on calculations of fission product release documented in NUREG/CR-5009 (16). NUREG/CR-5009 verifies the validity of using the Regulatory Guide

1.25 gap fractions when analyzing a fuel handling accident with extended burnup fuel of up to 60,000 MWD/MTU. However, Reference 16 bases this conclusion upon calculations which are designed to bound the possible releases from a fuel handling accident based on the single peak fuel rod. It is stated in NUREG/CR-5009 that the calculations are based on "the release from the peak operating rod in a fuel batch of a fuel design with high operating powers." Section 2.2 of NUREG/CR-5009 makes the following statement about the fuel gap fraction of the peak fuel rod versus the fuel gap fractions of the rest of the fuel rods in the core:

> *"It should be noted that the fission-product release from the peak operating rod in any given reactor core will be substantially greater than those from 95 to 99% of the fuel rods in a fuel batch at extended burnup. For example, 95 to 99% of the rods in any given fuel batch at a batch average burnup of 50 GWd/t have fission-gas (noble) release fractions between 0.015 and 0.025 (Pati and Garde 1985), whereas the calculated peak rod in the batch may have release fractions two to five times this amount. At the current batch average burnup of 33 GWd/t the majority of the rods (95% or more) in a fuel batch with an extended burnup fuel design will have release fractions less than or equal to 0.01, with the calculated peak rod having release fractions three to five times this amount."*

The relevant population of rods for the FHA event has been indicated as the rods in one fuel assembly – not the peak fuel rod. In addition, it was established that an assembly discharged from its first cycle of irradiation contained an isotopic inventory of iodine that would maximize the calculated dose. The gap fraction that is appropriate to characterize the population of rods in an entire fuel assembly (for burnup equivalent to one irradiation cycle) is approximately 0.01 as indicated in the citation from Reference 16 above. Similar results have been obtained from vendor calculations for fuel designs applicable to Surry cores. These calculations have concluded that for isotopes with half-lives of less than one year, the gap fraction is more sensitive to fuel temperature than to burnup. The fuel temperature depends upon the local power level in the fuel rods. All isotopes of radiological significance for a fuel handling accident have half-lives less than one year and are correctly characterized by this conclusion. Thus, the lower fuel temperatures associated with lower achievable peaking factors tend to decrease the gap fraction, when one accounts for inherent physical features in reactor cores.

The foregoing discussion has been provided as justification for application of the gap release fraction proposed in NUREG-1465 (1). The recommendation is repeated here as stated in Section 3.6 of NUREG-1465:

> *"1. Accidents where long-term fuel cooling is maintained despite fuel failure. Examples include the design basis LOCA where ECCS functions and a postulated spent fuel handling accident. For this category, fuel failure is taken to result in an immediate release, based upon Reference 5 and 16, of 3 percent of the volatile fission products (noble gases, iodine, and cesium) which are in the gap between the fuel pellet and the cladding. No subsequent appreciable release from the fuel pellet occurs, since the fuel does not experience prolonged high temperatures."*

Release Mechanism (i.e., cladding breach)

For the FHA event, no specific mechanism is postulated for cladding failure other than that already stated at the beginning of this section. It is assumed that all fuel rods in one assembly are damaged. This assumption, in conjunction with the quantified inventory in the rod/cladding gap, defines the total released material for the FHA radiological analysis. For other non-LOCA events, it would be necessary to determine the expected failure mechanisms, the extent of predicted failure and the type of rods in which failure may occur. Each of these factors would impact the total source term that would be released into the reactor coolant system for a given event.

Table 3.2-2 provides the total assumed activity in the fuel rod gap for each of the isotopes analyzed in the FHA event. This activity is the number of curies available for release from the failure of the cladding from all the rods in one fuel assembly, at the end of its first cycle of irradiation, assuming the gap fractions indicated for the FHA event in Table 3.2-1. This represents the assumed activity that is released to the water surrounding the failed fuel assembly.

Table 3.2-2 – Single Fuel Assembly Gap Inventory by Isotope (for FHA Analysis)

Isotope	Activity (Curies)
I-130	1.716E+00
I-131	1.502E+04
I-132	1.268E+04
I-133	1.590E+03
I-135	1.136E+00
Kr-85	1.162E+03
Kr-88	4.151E-07
Kr-83m	3.279E-09
Kr-85m	1.177E-03
Xe-133	3.007E+04
Xe-135	5.473E+01
Xe-131m	2.253E+02
Xe-133m	5.545E+02
Xe-135m	1.820E-01

3.2.3 Determination of Onsite Atmospheric Dispersion Factors (X/Q)

The onsite (Main Control Room) atmospheric dispersion factors that were employed in the FHA analysis are presented in Table 3.2-3. The source points modeled were the Unit 1 Containment building and Ventilation Vent No. 2, but the containment source has multiple potential release paths. The containment is exhausted through a purge system that has forced exhaust through Ventilation Vent No. 2. This analysis also accommodates potential releases from the equipment access hatch, the personnel airlock and penetrations that terminate in the Auxiliary Building or Safeguards. Ventilation Vent No. 2 is modeled since this is the discharge point for exhaust from the fuel building and containment purge. The equipment access hatch was modeled as a release directly from containment. The personnel airlock was modeled as a release from the Auxiliary Building 45 ft elevation east and west louvers, since this represents the likely pathway for these releases. The receptor points modeled were the turbine building fresh air louvers, the turbine building fresh air intakes, the turbine building rollup doors, the EAB and the LPZ.

3.2.4 FHA Analysis Assumptions & Key Parameter Values

3.2.4.1 Special Modeling Considerations

As in the LOCA analysis modeling, specific system features and traditional assumptions warranted special modeling attention. This allowed more appropriate representation of physical phenomena for use in the Surry FHA radiological analysis. Two such items are discussed in this section: 1) treatment of flowrates for the FHA analysis release paths; 2) selection of atmospheric dispersion factors to represent releases from the fuel building or containment purge (both via Ventilation Vent No. 2), the personnel airlock or equipment access hatch. These items are discussed in the sections that follow.

Effluent Flowrates Assumed for FHA Release Paths

In Appendix B of DG-1081 (3), it is stated that for FHA analyses, the radioactive material that escapes from the fuel pool or reactor cavity pool is released to the environment over a 2-hour time period. This requirement, which also appears in Regulatory Guide 1.25, has been previously implemented in existing Surry FHA analyses by artificially selecting an effluent flowrate that resulted in complete evacuation of all the radioactive material within 2 hours. This assumption has no relationship to actual plant ventilation system capability or other mechanisms, such as natural circulation, that may be present for specific FHA scenarios. In addition, sensitivity analyses performed during the AST implementation indicated that non-conservative control room doses may be obtained by applying the flowrates corresponding to total release within 2 hours. Therefore, the approach taken in the AST implementation analysis involves bounding the potential range of effluent flowrates that correspond to expected equipment capability or natural circulation flow processes. Any restriction to the air flow through the equipment hatch – such as curtains – is accommodated by the analysis. The analysis accommodates all credible modes of operation for the containment ventilation equipment and establishes no restrictions on its use. The flowrates assumed bound the credible range of sustained flowrates that may exist for effluents through either the fuel building or containment purge exhaust (via Ventilation Vent No. 2), the equipment access hatch, the personnel airlock or other open penetrations. The analysis results are applicable for FHA scenarios in which any one or any combination of these release pathways are open to the environment. The range of flowrates considered is indicated on Table 3.2-3.

X/Q Selection for Multiple Containment Release Paths

The X/Q values used to model various release pathways are presented in Table 3.2-3. A range of flowrates is assumed for containment releases in order to bound the potential release rates. Furthermore, the FHA analysis involved use of X/Q values that bound the effects for release from any of the potentially open pathways. Considerations in selecting the X/Q value included factors such as building wake effects, dilution and holdup effects and relative location of the potential openings in containment. This approach ensures that calculated doses are conservative for the proposed operation.

Table 3.2-3 summarizes analysis assumptions and key input parameter values that are unique to the FHA analysis cases.

Table 3.2-3

Analysis Assumptions & Key Parameter Values
Employed Only in Fuel Handling Accident Analysis

Containment Parameters

Release Flowrate (0 – 720 hours)	1000 – 36,000 cfm[1]
Free Volume (for holdup; 50% of total)	9.315E5 ft^3

Core and Fuel Assembly Characteristics

Number of Fuel Assemblies in Core	157
Maximum Fuel Assembly Radial Peaking Factor	1.62
Assumed Iodine Physical Form In Gap	99.75% elemental 0.25% organic

MCR Atmospheric Dispersion Factors

	Equipment Hatch	Personnel Airlock	Fuel Building/Purge
0 – 8 hour	1.69E-3 sec/m^3	3.87E-3 sec/m^3	1.60E-3 sec/m^3
8 – 24 hours	7.19E-4 sec/m^3	1.65E-3 sec/m^3	6.99E-4 sec/m^3
24 – 96 hours	1.66E-4 sec/m^3	1.14E-3 sec/m^3	1.71E-4 sec/m^3
96 – 720 hours	1.20E-4 sec/m^3	7.96E-4 sec/m^3	1.22E-4 sec/m^3

Miscellaneous

Decontamination Factor – Elemental Iodine	500
Decontamination Factor – Organic Iodine	1
Minimum Depth of Water Over Fuel	23 feet
Fuel Building Free Volume (for holdup)	1.11E5 ft^3
Fuel Building Release Flowrate (0 – 720 hours)	36,000 - 80,000 cfm[1]

Key Operator Actions	Timing of Action
Discharge Air Bottles/Isolate MCR Upon Indication of FHA	Prior to MCR Intake of Contaminated Air

[1] Release flowrates are assumed to be constant for the duration of the event. Dose consequences bound expected results from all credible flow combinations.

3.2.5 FHA Analysis Results

The results of the FHA dose analysis are presented in Table 3.2-4. These results report the calculated dose for the worst 2-hour interval (EAB), and for the assumed 30 day duration of the event for the control room and LPZ. The doses are calculated with the TEDE methodology, and are compared with the applicable acceptance criteria specified in 10 CFR 50.67 and DG-1081. It can be observed from the table that each of the results meets the dose acceptance criteria.

Table 3.2-4 – Fuel Handling Accident Analysis Results

Accident Location[1] & Release Path	Control Room Dose (rem TEDE)	EAB Dose (rem TEDE)	LPZ Dose (rem TEDE)
Containment Purge(Ventilation Vent No. 2) Personnel Airlock Equipment Hatch Penetrations Fuel Building (Ventilation Vent No. 2)	0.53	3.40	0.17
Acceptance Criteria	5.0	6.25	6.25

[1] Reported results are from limiting case(s) that bound(s) the consequences from each path listed

3.3 Evaluation of Unaffected Events

This section documents an evaluation of the impact of implementing the AST, including the proposed plant and Technical Specifications changes, upon radiological analyses that are documented in the Surry UFSAR. Documented below is the evaluation performed for the four remaining events having significant radiological consequences that are presented in the Surry UFSAR.

3.3.1 Steam Generator Tube Rupture

The radiological effects of a postulated steam generator tube rupture are documented in Surry UFSAR Section 14.3.1.4. The analyses are performed with the Westinghouse Owners' Group methodology (7) that incorporates the effects of potential SG tube uncovery during the event. In accordance with that methodology, the calculational model includes the tube uncovery effects through these two mechanisms, which dominate the dose results:

1) releases from secondary liquid boiling including allowance for a partition factor of 0.01 for iodine between secondary liquid and steam.

2) releases from the fraction of primary liquid break flow that flashes to steam. A partition factor of 1 is assumed for this flashing fraction.

The analysis has been performed assuming cases with both a pre-accident and concurrent iodine spike, in accordance with guidance in NUREG-0800, Section 15.6.3 (15). The thermal-hydraulic analysis of the SGTR accidents indicate that no fuel rod failures occur as a result of this transient. Thus, radioactive material releases are determined by the radionuclide concentrations initially present in primary liquid, secondary liquid and secondary steam, plus any releases from fuel rods that have failed before the transient. Both a pre-accident iodine spike and a concurrent accident iodine spike were modeled, in conjunction with the applicable Technical Specifications limit on reactor coolant activity in each case. These limits on iodine concentration are unaffected by implementation of the AST. For the case of a concurrent iodine spike, the UFSAR analysis assumes that iodine release from failed fuel rods is at a rate 500 times the release rate corresponding to the Technical Specifications limit for normal operations. The SGTR results presented in UFSAR Section 14.3.1.4 are thus unaffected, and remain acceptable for operation following implementation of the AST for Surry Units 1 and 2.

3.3.2 Main Steamline Break

The radiological effects of a postulated main steamline break are documented in Surry UFSAR Section 14.3.2.4. The analyses are performed with assumptions concerning iodine source terms and releases as specified in Section 15.1.5 of NUREG-0800 (15). For the MSLB, the radioactive material releases are determined by the initial radionuclide concentrations present in primary liquid, secondary liquid and secondary steam, plus any releases from failed fuel rods (if predicted). The thermal-hydraulic analysis for the MSLB predicts no fuel rod failures, so this additional source is not assumed.

The amount of activity in the primary and secondary coolant at the initiation of the MSLB is assumed to be at the maximum levels allowed by the plant Technical Specifications. Both a pre-accident iodine spike and a concurrent accident iodine spike were modeled, in conjunction with the applicable Technical Specifications limit on reactor coolant activity in each case. These limits on iodine concentration are unaffected by implementation of the AST. For the case of a concurrent iodine spike, the UFSAR analysis assumes that iodine release from failed fuel rods is at a rate 500 times the release rate corresponding to the Technical Specifications limit for normal operations. The Main Steamline Break results presented in UFSAR Section 14.3.2.4 thus remain acceptable for operation following implementation of the AST for Surry Units 1 and 2.

3.3.3 Locked Rotor

The radiological effects of a postulated locked reactor coolant pump rotor are documented in Surry UFSAR Section 14.2.9.2.4. The analysis accounts for release of radioactivity from primary and secondary side coolant, via primary-to-secondary leakage, and from fission product releases associated with postulated failed fuel rods that occur during the event. The amount of activity in the primary and secondary coolant at the initiation of the accident is assumed to be at the maximum levels allowed by the plant Technical Specifications. The primary coolant activity level also assumes a pre-accident iodine spike to the maximum level allowed by the Surry Technical Specifications. These limits on iodine concentration are unaffected by implementation of the AST. The analysis assumes an additional source from the release of fission products in 5% of the core fuel rods, in which the cladding is assumed to fail during the event. This assumption is conservative, since the existing thermal-hydraulic analysis for the locked rotor concludes that no rods fail. The locked rotor results presented in UFSAR Section 14.2.9.2.4 thus remain acceptable for operation following implementation of the AST for Surry Units 1 and 2.

3.3.4 Volume Control Tank Rupture

The radiological effects of this event are documented in UFSAR Section 14.4.2.1. The calculated doses are dependent upon the total curies contained in the tank and letdown flowrate, and are based on reactor coolant equilibrium activities with 1% failed fuel. This total activity is derived from operational considerations that are not affected by the postulated accident source term defined in NUREG-1465. The volume control tank rupture results presented in the UFSAR thus remain acceptable for operation following implementation of the AST for Surry Units 1 and 2.

3.3.5 Waste Gas Decay Tank Rupture

The radiological effects of this event are documented in UFSAR Section 14.4.2.1. The calculated doses are dependent upon the total limit on activity contained in the tank, which is specified in Technical Specifications. This activity is itself derived from operational considerations that are not affected by the postulated accident source term defined in NUREG-1465. The waste gas decay tank rupture results presented in the UFSAR thus remain acceptable for operation following implementation of the AST for Surry Units 1 and 2.

4.0 Additional Design Basis Considerations

In addition to the explicit evaluation of radiological consequences that had direct impact from the changes associated with implementing the AST, other areas of plant design were also considered for potential impacts. The evaluation of these additional design areas is documented below.

4.1 Impact Upon Equipment Environmental Qualification

The NRC, in its rebaselining study of AST impact (17), considered the effects of the AST on analyses of the postulated integrated radiation doses for plant components exposed to containment atmosphere radiation sources and those exposed to containment sump radiation sources. The NRC study concluded that the increased concentration of cesium in the containment sump water could result in an increase in the postulated integrated doses for certain plant components subject to equipment qualification. The increased cesium concentration in the source term causes (beyond a specific timeframe) the calculated integrated sump doses for the NUREG-1465 source term to exceed the doses based upon the TID-14844 source term. The Reference 17 analyses indicated that the timeframe at which the doses based upon the TID-14844 source term may be exceeded and become non-conservative is from approximately 7 to 30 days after the postulated LOCA, depending upon plant-specific assumptions and features.

The NRC sponsored a study, documented in NUREG/CR-5313 (18), to assess the impact of electrical equipment environmental qualification or lack thereof on reactor risk. This study evaluated the equipment that must function in various accident sequences, and determined the impact upon plant risk if such equipment were to fail (e.g., from exposure to harsh conditions beyond those for which it was qualified). The study concluded that equipment functions have high risk significance only if the equipment operation occurs during the first few days after accident initiation. The EQ issue associated with the AST is that there is a potential for integrated doses to exceed that for which equipment was qualified, but only for timeframes beyond 7 days. From the Reference (18) study, it is reasonable to conclude that this issue has low risk impact.

In the Federal Register notice issuing the final rule for use of alternative source terms at operating reactors (2), the NRC stated that it will evaluate this issue as a generic safety issue to determine whether further regulatory actions are justified. The notice also stated the NRC intent that the final regulatory guide (i.e., DG-1081) or subsequent revisions thereto, is expected to reflect the resolution of this generic safety issue. Further guidance is provided in SECY-99-240 (19), which transmitted the final AST rule changes for the Commission's approval. The following is stated in the 'Discussion' section, regarding evaluation of the equipment qualification issue before its final resolution:

> *"In the interim period before final resolution of this issue, the staff will consider the TID-14844 source term to be acceptable in reanalyses of the impact of proposed plant modifications on previously analyzed integrated component doses regardless of the accident source term used to evaluate offsite and control room doses."*

Consistent with this guidance, no further evaluation of this issue is presented in support of implementing the AST for Surry Units 1 and 2. The existing equipment qualification analyses, which are based upon the TID-14844 source term, are considered acceptable.

4.2 Risk Impact of Proposed Changes Associated with AST Implementation

Implementation of ASTs is of benefit to licensees because of the potential to obtain relaxation in specific safeguards systems operability or surveillance requirements, since such changes can reduce regulatory burden and streamline operations. Such changes are warranted if they can be pursued without creating an unacceptable impact upon plant risk characteristics as compared with the existing system licensing and operational basis. The proposed changes associated with implementation of the AST for Surry Units 1 and 2 have been considered for their risk effects. A discussion of these considerations is presented below.

The proposed changes are presented here for convenience along with the report section describing each:

- Open Personnel Air Lock, Equipment Hatch & Penetrations During Refueling (Section 2.2)
- Eliminate Filtration of Containment & Fuel Building Exhaust During Refueling (Section 2.3)
- Redefinition of Subatmospheric Containment Depressurization Criterion (Section 2.4)

The proposed change to allow the personnel airlock and/or equipment access hatch and certain penetrations to be open during refueling will not be applicable during power operation. This change thus has no effect upon plant risk and mitigation of incidents occurring during power operation. The potential impact is upon incidents that are postulated during shutdown that would be negatively affected by a temporary loss of containment integrity. The breach in containment is temporary since the proposed Technical Specifications changes require that the containment openings be capable of being closed. Changes will also be made to plant procedures to ensure that these openings are capable of being closed. In the case of the equipment access hatch, the duration of the containment opening will be dependent upon the severity of the fuel handling accident. Closure of the equipment hatch will be accomplished only as allowed by containment dose rates. This approach is itself the result of a risk judgement, in which it is deemed preferable to avoid the likely personnel hazard associated with prompt hatch closure, in exchange for the offsite exposure that may result from delaying closure. This tradeoff is deemed acceptable and is considered to cause a negligible change in the plant risk.

The current requirements to continuously filter the exhaust of the containment and fuel building during fuel handling activities are being eliminated. In addition, the LOCA analysis does not credit filtration of the iodine releases from ECCS leakage. The risk associated with modification and/or elimination of such filtration systems was evaluated during the rebaselining study. Reference (17) reported that the effect on overall risk from filtration system modifications was small. This effect was attributed to the fact that filtration systems, which require electrical power for operation, will already not be functional for certain risk-significant accident sequences (e.g., station blackout). In addition, the most risk-significant accident sequences involve containment bypass scenarios, for which filtration systems are ineffective. The proposed changes to eliminate credit for filtration are expected to produce negligible incremental change in overall plant risk in such sequences.

The proposed change to allow a short duration of slightly atmospheric containment conditions beyond the current one hour timeframe following the design basis LOCA is in effect an increase in the containment leak rate. Reference (17) evaluated the impact of a change in containment leak rate upon plant risk. It was concluded that plant risk was not very sensitive to such a change since risk is dominated by accident sequences that result in early containment failure or bypass of containment. The same conclusion is reached, in which there is negligible effect upon overall plant risk from the proposed operation.

It is concluded that the proposed changes associated with AST implementation for Surry Units 1 and 2 will have insignificant effect upon the risk associated with severe accidents. This is primarily due to the fact that the risk significant accident sequences involve the failure of systems or structures (e.g., containment) that are not impacted by the relatively minor operational changes proposed herein.

4.3 Impact Upon Emergency Planning Radiological Assessment Methodology

This application of the AST for Surry replaces the existing design basis source term with the source term defined in NUREG-1465. The MIDAS model that is employed for emergency planning radiological assessments includes definitions of source terms for various design basis accidents. Such calculation results from MIDAS are used in various emergency preparedness processes. The basis of the existing source term definitions in the MIDAS calculations will be evaluated to determine: 1) the manner in which the source terms used in emergency preparedness activities rely upon the design basis event source term definition and 2) what specific changes may be warranted in the emergency preparedness source terms and their detailed usage.

5.0 Conclusions

The alternative source term defined in NUREG-1465 and associated analysis guidance provided in DG-1081 has been incorporated into the reanalysis of radiological effects from two key accidents for Surry Units 1 and 2. This represents a full implementation of the alternative source term in which the NUREG-1465 source term will become the licensing basis source term for assessment of design basis events. The analysis results from the reanalyzed event meet all of the acceptance criteria as specified in 10 CFR 50.67 and DG-1081.

6.0 References

1. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1995.

2. "Use of Alternative Source Terms at Operating Reactors," Final Rule, in Federal Register No. 64, p. 71990, December 23, 1999.

3. Draft Regulatory Guide DG-1081, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," USNRC, Office of Nuclear Regulatory Research, December 1999.

4. LOCADOSE, Bechtel Standard Computer Program NE319, Version 4.1 (User's and Theoretical Manuals), Revision 4.

5. Letter, James P. O' Hanlon to USNRC, "Virginia Electric & Power Company, Surry Power Station Units 1 and 2, Proposed Technical Specifications Changes to Accommodate Core Uprating," Serial No. 94-509, August 30, 1994.

6. Letter, Bart C. Buckley (NRC) to James P. O' Hanlon, "Surry Units 1 and 2 - Issuance of Amendments Re: Uprated Core Power (Serial No. 94-509) (TAC Nos. M90364 and M90365)," August 4, 1995.

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8. Technical Information Document, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," USAEC, 1962.

9. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

10. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", EPA 520/1-88-020, Environment Protection Agency, 1988.

11. Federal Guidance Report No. 12, "External Exposures to Radionuclides in Air, Water and Soil", EPA 420-r-93-081, Environmental Protection Agency, 1993.

7.0 References (continued)

12. NUREG/CR-6331, Rev. 1, "Atmospheric Relative Concentrations in Building Wakes, ARCON96," USNRC, 1997.

13. NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," USNRC, 1982.

14. NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," USNRC, June 1993.

15. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," USNRC, Rev 2, December 1988.

16. NUREG/CR-5009, PNL-6258, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February, 1988.

17. SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," June 30, 1998.

18. NUREG/CR-5313, "Equipment Qualification (EQ) Risk Scoping Study," January 1989.

19. SECY-99-240, "Final Amendment to 10 CFR Parts 21, 50, and 54 and Availability for Public Comment of Draft Regulatory Guide DG-1081 and Draft Standard Review Plan Section 15.0.1 Regarding Use of Alternative Source Terms at Operating Reactors", October 5, 1999.

20. Letter, James P. O' Hanlon to USNRC, "Revised Plan For Alternate Source Term Implementation-Surry Power Station Units 1 And 2," Serial No. 99-620, January 10, 2000.

21. WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel", December 1985, Edited by S. L. Davidson and W .R. Kramer.

22. Safety Evaluation by the Division of Reactor Licensing, US Atomic Energy Commission, Surry Power Station Units 1 and 2, Docket Nos. 50-280 and 50-281, February 23, 1972.

Specific Changes

Revise the following current Technical Specifications for Units 1 and 2 as noted below to reflect implementation of the NUREG-1465 alternative source term (AST) as the Design Basis Source Term. The AST implementation analyses provide justification for the following changes to the Surry Technical Specifications: redefinition of the subatmospheric containment depressurization criterion; open personnel air lock and equipment access hatch during refueling; certain additional open penetrations during refueling; elimination of the containment purge isolation operability requirement during refueling; and elimination of requirements to filter containment and fuel building exhaust during fuel handling. An additional specification (TS 3.10.B) is added that is applicable for irradiated fuel movement in the Fuel Building. In this section deleted text is omitted and inserted text is underlined in the **To** portion of each revision.

Spray Systems

Revise the current Technical Specification 3.4 Basis from:

With one Containment Spray Subsystem and two Recirculation Spray Subsystems operating together, the spray systems are capable of cooling and depressurizing the containment to subatmospheric pressure in less than 60 minutes following the Design Basis Accident. The Recirculation Spray Subsystems are capable of maintaining subatmospheric pressure in the containment indefinitely following the Design Basis Accident when used in conjunction with the Containment Vacuum System to remove any long term air in leakage.

To

With one Containment Spray Subsystem and two Recirculation Spray Subsystems operating together, the spray systems are capable of cooling and depressurizing the containment to <u>0.5 psig</u> in less than 60 minutes <u>and to subatmospheric pressure within 4 hours</u> following the Design Basis Accident. The Recirculation Spray Subsystems are capable of maintaining subatmospheric pressure in the containment indefinitely following the Design Basis Accident when used in conjunction with the Containment Vacuum System to remove any long term air <u>inleakage. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig (from 1 hour to 4 hours) and is maintained less than 0.0 psig (after 4 hours).</u>

Containment

Revise current Technical Specification 3.8 Basis from:

If the containment air partial pressure rises to a point above the allowable value the reactor shall be brought to the HOT SHUTDOWN condition. If a LOCA occurs at the time the containment air partial pressure is at the maximum allowable value, the maximum containment pressure will be less than design pressure (45 psig), the containment will depressurize in less than 1 hour, and the maximum subatmospheric peak pressure will be less than 0.0 psig.

To

If the containment air partial pressure rises to a point above the allowable value the reactor shall be brought to the HOT SHUTDOWN condition. If a LOCA occurs at the time the containment air partial pressure is at the maximum allowable value, the maximum containment pressure will be less than design pressure (45 psig), the containment will depressurize to 0.5 psig within 1 hour and less than 0.0 psig within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident.

Refueling

Revise current Technical Specification 3.10 Applicability from:

Applicability
Applies to operating limitations during REFUELING OPERATIONS.

To

Applicability
Applies to operating limitations during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building.

Revise current Technical Specification 3.10 Objective from:

Objective
To assure that no accident could occur during REFUELING OPERATIONS that would affect public health and safety.

To

Objective
To assure that no accident could occur during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building that would affect public health and safety.

Revise current Technical Specification 3.10.A.1 from:

A. During REFUELING OPERATIONS the following conditions are satisfied:
   1. The equipment access hatch and at least one door in the personnel airlock shall be properly closed. For those penetrations which provide a direct path from containment atmosphere to the outside atmosphere, the automatic containment isolation valves shall be operable or the penetration shall be closed by a valve, blind flange, or equivalent.

To

A. During REFUELING OPERATIONS the following conditions are satisfied:
   1. The equipment access hatch and at least one door in the personnel airlock shall be capable of being closed. For those penetrations which provide a direct path from containment atmosphere to the outside atmosphere, the containment isolation valves shall be OPERABLE or the penetration shall be closed by a valve, blind flange, or equivalent or the penetration shall be capable of being closed.

Delete current Technical Specification 3.10.A.2:

2. The Containment Ventilation Purge System and the area and airborne radiation monitors which initiate isolation of this system shall be tested and verified to be operable immediately prior to REFUELING OPERATIONS.

Revise current Technical Specification 3.10.A.4 from:

4. Manipulator crane area radiation levels and airborne activity levels within the containment and airborne activity levels in the ventilation exhaust duct shall be continuously monitored during refueling. A manipulator crane high radiation alarm or high airborne activity level alarm within the containment will automatically stop the purge ventilation fans and automatically close the containment purge isolation valves.

To

4. The manipulator crane area monitors and the containment particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

Delete current Technical Specification 3.10.A.5:

5. Fuel pit bridge area radiation levels and ventilation vent exhaust airborne activity levels shall be continuously monitored during refueling. The fuel building exhaust will be continuously bypassed through the iodine filter bank during refueling procedures, prior to discharge through the ventilation vent.

Delete current Technical Specification 3.10.A.13:

13. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.

Revise current Technical Specification 3.10.A.14 from:

14. Two trains of the control and relay room emergency ventilation system shall be operable. With one train inoperable for any reason, demonstrate the other train is operable by

performing the test in Specification 4.20.A.1. With both trains inoperable, comply with Specification 3.10.B.

To

14. Two trains of the control and relay room emergency ventilation system shall be OPERABLE. With one train inoperable for any reason, demonstrate the other train is OPERABLE by performing the test in Specification 4.20.A.1. With both trains inoperable, comply with Specification 3.10.C.

Revise current Technical Specification 3.10.A.15 from:

15. Containment purge shall be filtered through high efficiency particulate air filters and charcoal absorbers.

To

15. Two trains of the control room bottled air system shall be OPERABLE. With one train inoperable for any reason, restore the inoperable train to OPERABLE status within 7 days or comply with Specification 3.10.C. With two trains inoperable, comply with Specification 3.10.C.

Due to the deletion of 3 of the 15 conditions required in TS 3.10.A the conditions were renumbered as 1 through 12. Additional editorial changes were made to capitalize the word OPERABLE throughout TS 3.10.A.

Add new Technical Specification 3.10.B:

B. During irradiated fuel movement in the Fuel Building the following conditions are satisfied:

1. The fuel pit bridge area monitor and the ventilation vent stack 2 particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

2. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

   This restriction does not apply to the movement of the transfer canal door.

3. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.

4. Two trains of the control and relay room emergency ventilation system shall be OPERABLE. With one train inoperable for any reason, demonstrate the other train

is OPERABLE by performing the test in Specification 4.20.A.1. With both trains inoperable, comply with Specification 3.10.C.

5. Two trains of the control room bottled air system shall be OPERABLE. With one train inoperable for any reason, restore the inoperable train to OPERABLE status within 7 days or comply with Specification 3.10.C. With two trains inoperable, comply with Specification 3.10.C.

Revise and Renumber current Technical Specification 3.10.B from:

B. If any one of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

To

C. If any one of the specified limiting conditions for refueling is not met, REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building shall cease, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

Renumber current Technical Specifications 3.10.C and 3.10.D as 3.10.D and 3.10.E respectively.

Revise current Technical Specification 3.10 Basis from:

Detailed instructions, the above specified precautions, and the design of the fuel handling equipment, which incorporates built-in interlocks and safety features, provide assurance that an accident, which would result in a hazard to public health and safety, will not occur during unit REFUELING OPERATIONS. When no change is being made in core geometry, one neutron detector is sufficient to monitor the core and permits maintenance of the out-of-function instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

To

Detailed instructions, the above specified precautions, and the design of the fuel handling equipment, which incorporates built-in interlocks and safety features, provide assurance that an accident, which would result in a hazard to public health and safety, will not occur during unit REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building. When no change is being made in core geometry, one neutron detector is sufficient to monitor the core and permits maintenance of the out-of-function instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

Revise current Technical Specification 3.10 Basis from:

The containment equipment access hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of the containment. During REFUELING OPERATIONS, the equipment hatch is held in place with at least four approximately equally spaced bolts.

The containment airlocks, which are also part of the containment pressure boundary, provide a means for personnel access during periods when CONTAINMENT INTEGRITY is required. Each airlock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors to remain open for extended periods when frequent containment entry is necessary. During REFUELING OPERATIONS, containment closure is required. Therefore, the door interlock mechanism may remain disabled, but one airlock door must remain closed. The emergency escape airlock (trunk) may be removed from the equipment access hatch during REFUELING OPERATIONS, provided the penetration is closed by an approved method which provides a temporary, atmospheric pressure ventilation barrier.

To

The containment equipment access hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of the containment. During REFUELING OPERATIONS, the equipment hatch must be capable of being closed.

The containment airlocks, which are also part of the containment pressure boundary, provide a means for personnel access during periods when CONTAINMENT INTEGRITY is required. Each airlock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors to remain open for extended periods when frequent containment entry is necessary. During REFUELING OPERATIONS, containment closure does not have to be maintained, but airlock doors may need to be closed to establish containment closure. Therefore, the door interlock mechanism may remain disabled, but one airlock door must be capable of being closed.

Revise current Technical Specification 3.10 Basis from:

Containment high radiation levels and high airborne activity levels automatically stop and isolate the Containment Ventilation Purge System. The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by at least one barrier during REFUELING OPERATIONS. Isolation may be achieved by an OPERABLE automatic isolation valve, a closed valve, a blind flange, or by an equivalent isolation method. Equivalent isolation methods must be evaluated and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier.

To

Containment penetrations that terminate in the Auxiliary Building or Safeguards and provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being closed by at least one barrier during REFUELING OPERATIONS. The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by at least one barrier during REFUELING OPERATIONS. Isolation may be achieved by an OPERABLE isolation valve, a closed valve, a blind flange, or by an equivalent isolation method. Equivalent isolation methods must be evaluated and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier.

For the personnel airlock, equipment access hatch, and other penetrations, 'capable of being closed' means the openings are able to be closed; they do not have to be sealed or meet the leakage criteria of TS 4.4. Station procedures exist that ensure in the event of a fuel handling accident, that the open personnel airlock and other penetrations can and will be closed. Closure of the equipment hatch will be accomplished in accordance with station procedures and as allowed by dose rates in containment. The radiological analysis of the fuel handling accident does not take credit for closure of the personnel airlock, equipment access hatch or other penetrations.

Revise current Technical Specification 3.10 Basis from:

The fuel building ventilation exhaust is diverted through charcoal filters whenever refueling is in progress. At least one flow path is required for cooling and mixing the coolant contained in the reactor vessel so as to maintain a uniform boron concentration and to remove residual heat.

To

The fuel building ventilation exhaust and containment ventilation purge exhaust may be diverted through charcoal filters whenever refueling is in progress. However, there is no requirement for filtration since the Fuel Handling Accident analysis takes no credit for these filters. At least one flow path is required for cooling and mixing the coolant contained in the reactor vessel so as to maintain a uniform boron concentration and to remove residual heat.

Revise current Technical Specification 3.10 Basis from:

The fuel handling accident has been analyzed based on the methodology outlined in Regulatory Guide 1.25. The analysis assumes 100% of the gap activity from the highest powered assembly is released after a 100-hour decay period following operation at 2605 MWt.

To

The fuel handling accident has been analyzed based on the methodology outlined in Draft Regulatory Guide DG-1081. The analysis assumes 100% of the gap activity from the highest powered assembly is released after a 100-hour decay period following operation at 2605 MWt.

Main Control Room Bottled Air System

Revise current Technical Specification 3.19 Basis from:

Following a design basis loss of coolant accident, the containment will be depressurized to subatmospheric condition in less than 1 hour; thus, terminating leakage from the containment. The main control room is maintained at a positive differential pressure using bottled air during the period when containment leakage may exist to prevent contamination.

To

Following a design basis loss of coolant accident, the containment will be depressurized to 0.5 psig in less than 1 hour and to subatmospheric pressure within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment. The main control room is maintained at a positive differential pressure using bottled air during the first hour, when the containment leakrate is greatest.

Auxiliary Ventilation Exhaust Filter Trains

Revise current Technical Specification 3.22 Applicability from:

Applies to the ability of the safety-related system to remove particulate matter and gaseous iodine following a LOCA or a refueling accident.

To

Applies to the ability of the safety-related system to remove particulate matter and gaseous iodine following a LOCA.

An editorial change was made to capitalize the word OPERABLE in current TS 3.22.A.

Revise current Technical Specification 3.22 Basis from:

The purpose of the filter trains located in the auxiliary building is to provide standby capability for removal of particulate and iodine contaminants from the exhaust air of the charging pump cubicles of the auxiliary building, fuel building, decontamination building, containment (during shutdown) and safeguards building adjacent to the containment which discharge through the ventilation vent and could require filtering prior to release. During normal plant operation, the exhaust from any one of these areas can be diverted, if required, through the auxiliary building filter trains remotely from the control room. The safeguards building exhaust and the charging pump cubicle exhaust are automatically diverted through the filter trains in the event of a LOCA (diverted on a safety injection system signal). The fuel building exhaust and purge exhaust are aligned to continuously pass through the filters during spent fuel handling.

When irradiated fuel is being handled, the system is manually placed in alignment to ensure the exhaust from the fuel handling areas passes through the filters. The automatic alignment feature of the ventilation system, which initiates on a safety injection signal, is defeated unless the has decayed for a sufficient period of time such that the radiological consequences of a fuel handling accident would be acceptable without iodine filtration. Defeating the automatic alignment feature requires that, in the event of a LOCA, manual actions be taken to realign the ventilation system to the charging pump cubicles and safeguards areas following actions to secure fuel handling activities.

To

The purpose of the filter trains located in the auxiliary building is to provide standby capability for removal of particulate and iodine contaminants from the exhaust air of the charging pump cubicles of the auxiliary building, fuel building, decontamination building, containment (during shutdown) and safeguards building adjacent to the containment which discharge through the ventilation vent and could require filtering prior to release. During normal plant operation, the exhaust from any one of these areas can be diverted, if required, through the auxiliary building filter trains remotely from the control room. The safeguards building exhaust and the charging pump cubicle exhaust are automatically diverted through the filter trains in the event of a LOCA (diverted on a safety injection system signal). The fuel building exhaust and purge exhaust are not required to be aligned to pass through the filters during spent fuel handling since the Fuel Handling Accident analysis takes no credit for these filters.

Relocate the following Setpoints and Functions from Technical Specification Table 3.7-5 to another Licensee Controlled Document:

Monitor Channel	Automatic Function At Alarm Conditions	Monitoring Requirements	Alarm Setpoint μCI/cc
2. Containment particulate and gas monitors (RM-RMS-159 & RM-RMS-160, RM-RMS-259 & RM-RMS-260)	Trips affected unit's purge supply fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)	See Specification 3.10	Particulate $\leq 9 \times 10^{-9}$ Gas $\leq 1 \times 10^{-5}$
3. Manipulator crane area monitors (RM-RMS-162 & RM-RMS-262)	Trips affected unit's purge supply fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)	See Specification 3.10	$\leq$ 50 mrem/hr

Attachment 2

Mark-up of Technical Specifications and Bases

Note: In the Technical Specifications mark-ups altered text has been indicated with a revision bar in the right margin. In addition to the revision bar, deleted text has been marked with a strikethrough and inserted text has been underlined.

Basis

The spray systems in each reactor unit consist of two separate parallel Containment Spray Subsystems, each of 100 percent capacity, and four separate parallel Recirculation Spray Subsystems, each of 50 percent capacity.

Each Containment Spray Subsystem draws water independently from the refueling water storage tank (RWST). The water in the tank is cooled to 45°F or below by circulating the water through one of the two RWST coolers with one of the two recirculating pumps. The water temperature is maintained by two mechanical refrigerating units as required. In each Containment Spray Subsystem, the water flows from the tank through an electric motor driven containment spray pump and is sprayed into the containment atmosphere through two separate sets of spray nozzles. The capacity of the spray systems to depressurize the containment in the event of a Design Basis Accident is a function of the pressure and temperature of the containment atmosphere, the service water temperature, and the temperature in the refueling water storage tank as discussed in the Basis of Specification 3.8.

Each Recirculation Spray Subsystem draws water from the common containment sump. In each subsystem the water flows through a recirculation spray pump and recirculation spray cooler, and is sprayed into the containment atmosphere through a separate set of spray nozzles. Two of the recirculation spray pumps are located inside the containment and two outside the containment in the containment auxiliary structure.

With one Containment Spray Subsystem and two Recirculation Spray Subsystems operating together, the spray systems are capable of cooling and depressurizing the containment to ~~subatmospheric pressure~~ 0.5 psig in less than 60 minutes and to subatmospheric pressure within 4 hours following the Design Basis Accident. The Recirculation Spray Subsystems are capable of maintaining subatmospheric pressure in the containment indefinitely following the Design Basis Accident when used in conjunction with the Containment Vacuum System to remove any long term air ~~in leakage~~ inleakage. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig (from 1 hour to 4 hours) and is maintained less than 0.0 psig (after 4 hours).

TABLE 3.7-5
AUTOMATIC FUNCTIONS
OPERATED FROM RADIATION MONITORS ALARM

Monitor Channel	Automatic Function At Alarm Conditions	Monitoring Requirements	Alarm Setpoint µCI/cc
1. Component cooling water radiation monitors	Shuts surge tank vent valve HCV-CC-100	See Specification 3.13	Twice Background
~~2. Containment particulate and gas monitors (RM-RMS-159 & RM-RMS-160, RM-RMS-259 & RM-RMS-260)~~	~~Trips affected unit's purge supply fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)~~	~~See Specificatoin 3.10~~	~~Particulate ≤ 9 x 10^{-9}~~ ~~Gas ≤ 1 x 10^{-5}~~
~~3. Manipulator crane area monitors (RM-RMS-162 & RM-RMS-262)~~	~~Trips affected unit's purge supply fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)~~	~~See Specification 3.10~~	~~≤ 50 mrem/hr~~

If the containment air partial pressure rises to a point above the allowable value the reactor shall be brought to the HOT SHUTDOWN condition. If a LOCA occurs at the time the containment air partial pressure is at the maximum allowable value, the maximum containment pressure will be less than design pressure (45 psig), the containment will depressurize ~~in less than 1 hour, and the maximum subatmospheric peak pressure will be less than 0.0 psig.~~ to 0.5 psig within 1 hour and less than 0.0 psig within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident.

If the containment air partial pressure cannot be maintained greater than or equal to 9.0 psia, the reactor shall be brought to the HOT SHUTDOWN condition. The shell and dome plate liner of the containment are capable of withstanding an internal pressure as low as 3 psia, and the bottom mat liner is capable of withstanding an internal pressure as low as 8 psia.

References

UFSAR Section 4.3.2	Reactor Coolant Pump
UFSAR Section 5.2	Containment Isolation
UFSAR Section 5.2.1	Design Bases
UFSAR Section 5.5.2	Isolation Design
UFSAR Section 6.3.2	Containment Vacuum System

3.10 REFUELING

Applicability

Applies to operating limitations during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building.

Objective

To assure that no accident could occur during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building that would affect public health and safety.

Specification

A. During REFUELING OPERATIONS the following conditions are satisfied:

1. The equipment access hatch and at least one door in the personnel airlock shall be ~~properly~~ capable of being closed. For those penetrations which provide a direct path from containment atmosphere to the outside atmosphere, the ~~automatic~~ containment isolation valves shall be OPERABLE or the penetration shall be closed by a valve, blind flange, or equivalent or the penetration shall be capable of being closed.

2. ~~The Containment Ventilation Purge System and the area and airborne radiation monitors which initiate isolation of this system shall be tested and verified to be operable immediately prior to REFUELING OPERATIONS.~~

2~~3~~. At least one source range neutron detector shall be in service at all times when the reactor vessel head is unbolted. Whenever core geometry or coolant chemistry is being changed, subcritical neutron flux shall be continuously monitored by at least two source range neutron detectors, each with continuous visual indication in the Main Control Room and one with audible indication within the containment. During core fuel loading phases, there shall be a minimum neutron count rate detectable on two operating source range neutron detectors with the exception of initial core loading, at which time a minimum neutron count rate need be established only when there are eight (8) or more fuel assemblies loaded into the reactor vessel.

3~~4~~. The manipulator crane area monitors and the containment particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident. ~~Manipulator crane area radiation levels and airborne activity levels within the containment and airborne activity levels in the ventilation exhaust duct shall be continuously monitored during refueling. A manipulator crane high radiation alarm or high airborne activity level alarm within the containment will automatically stop the purge ventilation fans and automatically close the containment purge isolation valves.~~

~~5. Fuel pit bridge area radiation levels and ventilation vent exhaust airborne activity levels shall be continuously monitored during refueling. The fuel building exhaust will be continuously bypassed through the iodine filter bank during refueling procedures, prior to discharge through the ventilation vent.~~

4~~6~~. At least one residual heat removal pump and heat exchanger shall be OPERABLE to circulate reactor coolant. The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of core alterations or reactor vessel surveillance inspections.

5~~7~~. Two residual heat removal pumps and heat exchangers shall be OPERABLE to circulate reactor coolant when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

6~~8.~~ At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange during movement of fuel assemblies.

7~~9~~. With the reactor vessel head unbolted or removed, any filled portions of the Reactor Coolant System and the refueling canal shall be maintained at a boron concentration which is:

a. Sufficient to maintain K-effective equal to 0.95 or less, and
b. Greater than or equal to 2300 ppm and shall be checked by sampling at least once every 72 hours.

8~~10~~. Direct communication between the Main Control Room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

9~~11.~~ No movement of irradiated fuel in the reactor core shall be accomplished until the reactor has been subcritical for a period of at least 100 hours.

10~~12~~. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

This restriction does not apply to the movement of the transfer canal door.

~~13. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.~~

11~~14~~. Two trains of the control and relay room emergency ventilation system shall be OPERABLE. With one train inoperable for any reason, demonstrate the other train is OPERABLE by performing the test in Specification 4.20.A.1. With both trains inoperable, comply with Specification 3.10.C~~B~~.

12~~15~~. ~~Containment purge shall be filtered through high efficiency particulate air filters and charcoal absorbers.~~ Two trains of the control room bottled air system shall be OPERABLE. With one train inoperable for any reason, restore the inoperable train to OPERABLE status within 7 days or comply with Specification 3.10.C. With two trains inoperable, comply with Specification 3.10.C.

B. During irradiated fuel movement in the Fuel Building the following conditions are satisfied:

1. The fuel pit bridge area monitor and the ventilation vent stack 2 particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

2. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

   This restriction does not apply to the movement of the transfer canal door.

3. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.

4. Two trains of the control and relay room emergency ventilation system shall be OPERABLE. With one train inoperable for any reason, demonstrate the other train is OPERABLE by performing the test in Specification 4.20.A.1. With both trains inoperable, comply with Specification 3.10.C.

5. Two trains of the control room bottled air system shall be OPERABLE. With one train inoperable for any reason, restore the inoperable train to OPERABLE status within 7 days or comply with Specification 3.10.C. With two trains inoperable, comply with Specification 3.10.C.

C~~B~~. If any one of the specified limiting conditions for refueling is not met, ~~refueling~~ REFUELING OPERATIONS ~~of the reactor~~ or irradiated fuel movement in the Fuel Building shall cease, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

D~~C~~. After initial fuel loading and after each core refueling operation and prior to reactor operation at greater than 75% of rated power, the movable incore detector system shall be utilized to verify proper power distribution.

E~~D~~. The requirements of 3.0.1 are not applicable.

Basis
Detailed instructions, the above specified precautions, and the design of the fuel handling equipment, which incorporates built-in interlocks and safety features, provide assurance that an accident, which would result in a hazard to public health and safety, will not occur during unit REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building. When no change is being made in core geometry, one neutron detector is sufficient to monitor the core and permits maintenance of the out-of-function instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

Potential escape paths for fission product radioactivity within containment are required to be closed or capable of closure to prevent the release to the environment. However, since there is no potential for significant containment pressurization during refueling, the Appendix J leakage criteria and tests are not applicable.

The containment equipment access hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of the containment. During REFUELING OPERATIONS, the equipment hatch ~~is held in place with at least four approximately equally spaced bolts~~ must be capable of being closed.

The containment airlocks, which are also part of the containment pressure boundary, provide a means for personnel access during periods when CONTAINMENT INTEGRITY is required. Each airlock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors to remain open for extended periods when frequent containment entry is necessary. During REFUELING OPERATIONS, containment closure ~~is required~~ does not have to be maintained, but airlock doors may need to be closed to establish containment closure. Therefore, the door interlock mechanism may remain disabled, but one airlock door must ~~remain~~ be capable of being closed~~. The emergency escape airlock (trunk) may be removed from the equipment access hatch during REFUELING OPERATIONS, provided the penetration is closed by an approved method which provides a temporary, atmospheric pressure ventilation barrier.~~

~~Containment high radiation levels and high airborne activity levels automatically stop and isolate the Containment Ventilation Purge System.~~ Containment penetrations that terminate in the Auxiliary Building or Safeguards and provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being closed by at least one barrier during REFUELING OPERATIONS. The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by at least one barrier during REFUELING OPERATIONS. Isolation may be achieved by an OPERABLE ~~automatic~~ isolation valve, a closed valve, a blind flange, or by an equivalent isolation method. Equivalent isolation methods must be evaluated and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier.

For the personnel airlock, equipment access hatch, and other penetrations, 'capable of being closed' means the openings are able to be closed; they do not have to be sealed or meet the leakage criteria of TS 4.4. Station procedures exist that ensure in the event of a fuel handling accident, that the open personnel airlock and other penetrations can and will be closed. Closure of the equipment hatch will be accomplished in accordance with station procedures and as allowed by dose rates in containment. The radiological analysis of the fuel handling accident does not take credit for closure of the personnel airlock, equipment access hatch or other penetrations.

The fuel building ventilation exhaust and containment ventilation purge exhaust may be ~~is~~ diverted through charcoal filters whenever refueling is in progress. However, there is no requirement for filtration since the Fuel Handling Accident analysis takes no credit for these filters. At least one flow path is required for cooling and mixing the coolant contained in the reactor vessel so as to maintain a uniform boron concentration and to remove residual heat.

During refueling, the reactor refueling water cavity is filled with approximately 220,000 gal of water borated to at least 2,300 ppm boron. The boron concentration of this water, established by Specification 3.10.A.9, is sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the COLD SHUTDOWN condition with all control rod assemblies inserted. This includes a 1% $\Delta k/k$ and a 50 ppm boron concentration allowance for uncertainty. This concentration is also sufficient to maintain the core subcritical with no control rod assemblies inserted into the reactor. Checks are performed during the reload design and safety analysis process to ensure the K-effective is equal to or less than 0.95 for each core. Periodic checks of refueling water boron concentration assure the proper shutdown margin. Specification 3.10.A.10 allows the Control Room Operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are used during refueling to assure safe handling of the fuel assemblies. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

Upon each completion of core loading and installation of the reactor vessel head, specific mechanical and electrical tests will be performed prior to initial criticality.

The fuel handling accident has been analyzed based on the methodology outlined in Draft Regulatory Guide DG-1081~~1.25~~. The analysis assumes 100% of the gap activity from the highest powered assembly is released after a 100-hour decay period following operation at 2605 MWt.

Detailed procedures and checks insure that fuel assemblies are loaded in the proper locations in the core. As an additional check, the movable incore detector system will be used to verify proper power distribution. This system is capable of revealing any assembly enrichment error or loading error which could cause power shapes to be peaked in excess of design value.

References

UFSAR Section 5.2	Containment Isolation
UFSAR Section 6.3	Consequence Limiting Safeguards
UFSAR Section 9.12	Fuel Handling System
UFSAR Section 11.3	Radiation Protection
UFSAR Section 13.3	Table 13.3-1
UFSAR Section 14.4.1	Fuel Handling Accidents
FSAR Supplement:	Volume I: Question 3.2

If the requirements of Specification 3.19.A are not met within 48 hours after achieving hot shutdown condition, the unit shall be placed in the cold shutdown condition.

Basis

Following a design basis loss of coolant accident, the containment will be depressurized to ~~subatmospheric condition~~ 0.5 psig in less than 1 hour and to subatmospheric pressure within 4 hours. ~~; thus, terminating leakage from the containment. The main control room is maintained at a positive differential pressure using bottled air during the period when containment leakage may exist to prevent contamination.~~ The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment. The main control room is maintained at a positive differential pressure using bottled air during the first hour, when the containment leakrate is greatest.

3.22 AUXILIARY VENTILATION EXHAUST FILTER TRAINS

Applicability

Applies to the ability of the safety-related system to remove particulate matter and gaseous iodine following a LOCA ~~or a refueling accident~~.

Objective

To specify requirements to ensure the proper function of the system.

Specification

A. Whenever either unit's Reactor Coolant System temperature and pressure is greater than 350°F and 450 psig, respectively, two auxiliary ventilation exhaust filter trains shall be OPERABLE with:

   1. Two filter exhaust fans;
   2. Two HEPA filter and charcoal adsorber assemblies.

B. With one train of the exhaust filter system inoperable for any reason, return the inoperable train to an operable status within 7 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 48 hours.

Basis

The purpose of the filter trains located in the auxiliary building is to provide standby capability for removal of particulate and iodine contaminants from the exhaust air of the charging pump cubicles of the auxiliary building, fuel building, decontamination building, containment (during shutdown) and safeguards building adjacent to the containment which discharge through the ventilation vent and could require filtering prior to release. During normal plant operation, the exhaust from any one of these areas can be diverted, if required, through the auxiliary building filter trains remotely from the control room. The safeguards building exhaust and the charging pump cubicle exhaust are automatically diverted through the filter trains in the event of a LOCA (diverted on a safety injection system signal). The fuel building exhaust and purge exhaust are not required to be aligned to ~~continuously~~ pass through the filters during spent fuel handling since the Fuel Handling Accident analysis takes no credit for these filters.

~~When irradiated fuel is being handled, the system is manually placed in alignment to ensure the exhaust from the fuel handling areas passes through the filters. The automatic alignment feature of the ventilation system, which initiates on a safety injection signal, is defeated unless the has decayed for a sufficient period of time such that the radiological consequences of a fuel handling accident would be acceptable without iodine filtration. Defeating the automatic alignment feature requires that, in the event of a LOCA, manual actions be taken to realign the ventilation system to the charging pump cubicles and safeguards areas following actions to secure fuel handling activities.~~

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

Corrected By Letter Dated November 20, 1992

Attachment 3

Proposed Technical Specifications and Bases Change

Basis

The spray systems in each reactor unit consist of two separate parallel Containment Spray Subsystems, each of 100 percent capacity, and four separate parallel Recirculation Spray Subsystems, each of 50 percent capacity.

Each Containment Spray Subsystem draws water independently from the refueling water storage tank (RWST). The water in the tank is cooled to 45°F or below by circulating the water through one of the two RWST coolers with one of the two recirculating pumps. The water temperature is maintained by two mechanical refrigerating units as required. In each Containment Spray Subsystem, the water flows from the tank through an electric motor driven containment spray pump and is sprayed into the containment atmosphere through two separate sets of spray nozzles. The capacity of the spray systems to depressurize the containment in the event of a Design Basis Accident is a function of the pressure and temperature of the containment atmosphere, the service water temperature, and the temperature in the refueling water storage tank as discussed in the Basis of Specification 3.8.

Each Recirculation Spray Subsystem draws water from the common containment sump. In each subsystem the water flows through a recirculation spray pump and recirculation spray cooler, and is sprayed into the containment atmosphere through a separate set of spray nozzles. Two of the recirculation spray pumps are located inside the containment and two outside the containment in the containment auxiliary structure.

With one Containment Spray Subsystem and two Recirculation Spray Subsystems operating together, the spray systems are capable of cooling and depressurizing the containment to 0.5 psig in less than 60 minutes and to subatmospheric pressure within 4 hours following the Design Basis Accident. The Recirculation Spray Subsystems are capable of maintaining subatmospheric pressure in the containment indefinitely following the Design Basis Accident when used in conjunction with the Containment Vacuum System to remove any long term air inleakage. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig (from 1 hour to 4 hours) and is maintained less than 0.0 psig (after 4 hours).

TABLE 3.7-5
AUTOMATIC FUNCTIONS
OPERATED FROM RADIATION MONITORS ALARM

Monitor Channel	Automatic Function At Alarm Conditions	Monitoring Requirements	Alarm Setpoint µ CI/cc
1. Component cooling water radiation monitors	Shuts surge tank vent valve HCV-CC-100	See Specification 3.13	Twice Background

If the containment air partial pressure rises to a point above the allowable value the reactor shall be brought to the HOT SHUTDOWN condition. If a LOCA occurs at the time the containment air partial pressure is at the maximum allowable value, the maximum containment pressure will be less than design pressure (45 psig), the containment will depressurize to 0.5 psig within 1 hour and less than 0.0 psig within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident.

If the containment air partial pressure cannot be maintained greater than or equal to 9.0 psia, the reactor shall be brought to the HOT SHUTDOWN condition. The shell and dome plate liner of the containment are capable of withstanding an internal pressure as low as 3 psia, and the bottom mat liner is capable of withstanding an internal pressure as low as 8 psia.

References

UFSAR Section 4.3.2	Reactor Coolant Pump
UFSAR Section 5.2	Containment Isolation
UFSAR Section 5.2.1	Design Bases
UFSAR Section 5.5.2	Isolation Design
UFSAR Section 6.3.2	Containment Vacuum System

3.10 REFUELING

Applicability

Applies to operating limitations during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building.

Objective

To assure that no accident could occur during REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building that would affect public health and safety.

Specification

A. During REFUELING OPERATIONS the following conditions are satisfied:

   1. The equipment access hatch and at least one door in the personnel airlock shall be capable of being closed. For those penetrations which provide a direct path from containment atmosphere to the outside atmosphere, the containment isolation valves shall be OPERABLE or the penetration shall be closed by a valve, blind flange, or equivalent or the penetration shall be capable of being closed.

2. At least one source range neutron detector shall be in service at all times when the reactor vessel head is unbolted. Whenever core geometry or coolant chemistry is being changed, subcritical neutron flux shall be continuously monitored by at least two source range neutron detectors, each with continuous visual indication in the Main Control Room and one with audible indication within the containment. During core fuel loading phases, there shall be a minimum neutron count rate detectable on two operating source range neutron detectors with the exception of initial core loading, at which time a minimum neutron count rate need be established only when there are eight (8) or more fuel assemblies loaded into the reactor vessel.

3. The manipulator crane area monitors and the containment particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

4. At least one residual heat removal pump and heat exchanger shall be OPERABLE to circulate reactor coolant. The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of core alterations or reactor vessel surveillance inspections.

5. Two residual heat removal pumps and heat exchangers shall be OPERABLE to circulate reactor coolant when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

6. At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange during movement of fuel assemblies.

7. With the reactor vessel head unbolted or removed, any filled portions of the Reactor Coolant System and the refueling canal shall be maintained at a boron concentration which is:

   a. Sufficient to maintain K-effective equal to 0.95 or less, and

   b. Greater than or equal to 2300 ppm and shall be checked by sampling at least once every 72 hours.

8. Direct communication between the Main Control Room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

9. No movement of irradiated fuel in the reactor core shall be accomplished until the reactor has been subcritical for a period of at least 100 hours.

10. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

    This restriction does not apply to the movement of the transfer canal door.

11. Two trains of the control and relay room emergency ventilation system shall be OPERABLE. With one train inoperable for any reason, demonstrate the other train is OPERABLE by performing the test in Specification 4.20.A.1. With both trains inoperable, comply with Specification 3.10.C.

12. Two trains of the control room bottled air system shall be OPERABLE. With one train inoperable for any reason, restore the inoperable train to OPERABLE status within 7 days or comply with Specification 3.10.C. With two trains inoperable, comply with Specification 3.10.C.

B. During irradiated fuel movement in the Fuel Building the following conditions are satisfied:

1. The fuel pit bridge area monitor and the ventilation vent stack 2 particulate and gas monitors shall be OPERABLE and continuously monitored to identify the occurrence of a fuel handling accident.

2. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

   This restriction does not apply to the movement of the transfer canal door.

3. A spent fuel cask shall not be moved into the Fuel Building unless the Cask Impact Pads are in place on the bottom of the spent fuel pool.

4. Two trains of the control and relay room emergency ventilation system shall be OPERABLE. With one train inoperable for any reason, demonstrate the other train is OPERABLE by performing the test in Specification 4.20.A.1. With both trains inoperable, comply with Specification 3.10.C.

5. Two trains of the control room bottled air system shall be OPERABLE. With one train inoperable for any reason, restore the inoperable train to OPERABLE status within 7 days or comply with Specification 3.10.C. With two trains inoperable, comply with Specification 3.10.C.

C. If any one of the specified limiting conditions for refueling is not met, REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building shall cease, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

D. After initial fuel loading and after each core refueling operation and prior to reactor operation at greater than 75% of rated power, the movable incore detector system shall be utilized to verify proper power distribution.

E. The requirements of 3.0.1 are not applicable.

Basis

Detailed instructions, the above specified precautions, and the design of the fuel handling equipment, which incorporates built-in interlocks and safety features, provide assurance that an accident, which would result in a hazard to public health and safety, will not occur during unit REFUELING OPERATIONS or irradiated fuel movement in the Fuel Building. When no change is being made in core geometry, one neutron detector is sufficient to monitor the core and permits maintenance of the out-of-function instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

Potential escape paths for fission product radioactivity within containment are required to be closed or capable of closure to prevent the release to the environment. However, since there is no potential for significant containment pressurization during refueling, the Appendix J leakage criteria and tests are not applicable.

The containment equipment access hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of the containment. During REFUELING OPERATIONS, the equipment hatch must be capable of being closed.

The containment airlocks, which are also part of the containment pressure boundary, provide a means for personnel access during periods when CONTAINMENT INTEGRITY is required. Each airlock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors to remain open for extended periods when frequent containment entry is necessary. During REFUELING OPERATIONS, containment closure does not have to be maintained, but airlock doors may need to be closed to establish containment closure. Therefore, the door interlock mechanism may remain disabled, but one airlock door must be capable of being closed.

Containment penetrations that terminate in the Auxiliary Building or Safeguards and provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being closed by at least one barrier during REFUELING OPERATIONS. The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by at least one barrier during REFUELING OPERATIONS. Isolation may be achieved by an OPERABLE isolation valve, a closed valve, a blind flange, or by an equivalent isolation method. Equivalent isolation methods must be evaluated and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier.

For the personnel airlock, equipment access hatch, and other penetrations, 'capable of being closed' means the openings are able to be closed; they do not have to be sealed or meet the leakage criteria of TS 4.4. Station procedures exist that ensure in the event of a fuel handling accident, that the open personnel airlock and other penetrations can and will be closed. Closure of the equipment hatch will be accomplished in accordance with station procedures and as allowed by dose rates in containment. The radiological analysis of the fuel handling accident does not take credit for closure of the personnel airlock, equipment access hatch or other penetrations.

The fuel building ventilation exhaust and containment ventilation purge exhaust may be diverted through charcoal filters whenever refueling is in progress. However, there is no requirement for filtration since the Fuel Handling Accident analysis takes no credit for these filters. At least one flow path is required for cooling and mixing the coolant contained in the reactor vessel so as to maintain a uniform boron concentration and to remove residual heat.

During refueling, the reactor refueling water cavity is filled with approximately 220,000 gal of water borated to at least 2,300 ppm boron. The boron concentration of this water, established by Specification 3.10.A.9, is sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the COLD SHUTDOWN condition with all control rod assemblies inserted. This includes a 1% $\Delta k/k$ and a 50 ppm boron concentration allowance for uncertainty. This concentration is also sufficient to maintain the core subcritical with no control rod assemblies inserted into the reactor. Checks are performed during the reload design and safety analysis process to ensure the K-effective is equal to or less than 0.95 for each core. Periodic checks of refueling water boron concentration assure the proper shutdown margin. Specification 3.10.A.10 allows the Control Room Operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are used during refueling to assure safe handling of the fuel assemblies. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

Upon each completion of core loading and installation of the reactor vessel head, specific mechanical and electrical tests will be performed prior to initial criticality.

The fuel handling accident has been analyzed based on the methodology outlined in Draft Regulatory Guide DG-1081. The analysis assumes 100% of the gap activity from the highest powered assembly is released after a 100-hour decay period following operation at 2605 MWt.

Detailed procedures and checks insure that fuel assemblies are loaded in the proper locations in the core. As an additional check, the movable incore detector system will be used to verify proper power distribution. This system is capable of revealing any assembly enrichment error or loading error which could cause power shapes to be peaked in excess of design value.

References

UFSAR Section 5.2	Containment Isolation
UFSAR Section 6.3	Consequence Limiting Safeguards
UFSAR Section 9.12	Fuel Handling System
UFSAR Section 11.3	Radiation Protection
UFSAR Section 13.3	Table 13.3-1
UFSAR Section 14.4.1	Fuel Handling Accidents
FSAR Supplement:	Volume I: Question 3.2

If the requirements of Specification 3.19.A are not met within 48 hours after achieving hot shutdown condition, the unit shall be placed in the cold shutdown condition.

Basis

Following a design basis loss of coolant accident, the containment will be depressurized to 0.5 psig in less than 1 hour and to subatmospheric pressure within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment. The main control room is maintained at a positive differential pressure using bottled air during the first hour, when the containment leakrate is greatest.

3.22 AUXILIARY VENTILATION EXHAUST FILTER TRAINS

Applicability

Applies to the ability of the safety-related system to remove particulate matter and gaseous iodine following a LOCA.

Objective

To specify requirements to ensure the proper function of the system.

Specification

A. Whenever either unit's Reactor Coolant System temperature and pressure is greater than 350°F and 450 psig, respectively, two auxiliary ventilation exhaust filter trains shall be OPERABLE with:

   1. Two filter exhaust fans;

   2. Two HEPA filter and charcoal adsorber assemblies.

B. With one train of the exhaust filter system inoperable for any reason, return the inoperable train to an operable status within 7 days or be in at least Hot Shutdown within the next 6 hours and in Cold Shutdown within the following 48 hours.

Basis

The purpose of the filter trains located in the auxiliary building is to provide standby capability for removal of particulate and iodine contaminants from the exhaust air of the charging pump cubicles of the auxiliary building, fuel building, decontamination building, containment (during shutdown) and safeguards building adjacent to the containment which discharge through the ventilation vent and could require filtering prior to release. During normal plant operation, the exhaust from any one of these areas can be diverted, if required, through the auxiliary building filter trains remotely from the control room. The safeguards building exhaust and the charging pump cubicle exhaust are automatically diverted through the filter trains in the event of a LOCA (diverted on a safety injection system signal). The fuel building exhaust and purge exhaust are not required to be aligned to pass through the filters during spent fuel handling since the Fuel Handling Accident analysis takes no credit for these filters.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

Corrected By Letter Dated November 20, 1992

Attachment 4

Significant Hazards Consideration Determination

Significant Hazards Consideration

The proposed TS changes allow relaxation of containment integrity requirements during refueling operations by allowing the personnel airlock, equipment access hatch and certain penetrations to remain open during fuel movement in containment. The changes also eliminate the requirement to filter the exhaust from containment or the fuel building during refueling operations. Also proposed is a relaxation of the current containment design basis acceptance criteria to allow an interval of four hours following the design basis LOCA until containment is depressurized to subatmospheric conditions. We have reviewed the proposed TS changes relative to the requirements of 10 CFR 50.92 and determined that a significant hazards consideration is not involved. Specifically, operation of Surry Power Station with the proposed changes will not:

1. **Involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The probability remains unaffected since the accident analyses involve no change to a system, component or structure that affects initiating events for any of the accidents evaluated. The consequences of the reanalyzed events is expressed in terms of the TEDE dose, which is not directly comparable to either the thyroid or whole body doses reported in existing analyses. However, even taking this comparison into consideration, any dose increase is not significant. Furthermore, the revised analysis results meet the applicable TEDE dose acceptance criteria for alternative source term implementation.

2. **Create the possibility of a new or different kind of accident from any accident previously evaluated.**

   The implementation of the proposed changes does not create the possibility of an accident of a different type than was previously evaluated in the SAR. The proposed Technical Specifications changes allow relaxation of these current requirements: 1) maintaining subatmospheric containment conditions following a LOCA; 2) filtration of containment & fuel building exhaust during fuel movement; 3) maintaining the personnel airlock, equipment access hatch & penetrations closed during fuel movement and 4) operability of containment purge isolation during refueling. These changes do not alter the nature of events postulated in the UFSAR nor do they introduce any unique precursor mechanisms. Therefore, there is no possibility for accidents of a different type than previously evaluated.

3. **Involve a significant reduction in the margin of safety.**

The implementation of the proposed changes does not reduce the margin of safety. The radiological analysis results, even though compared with the revised TEDE acceptance criteria, meet the applicable limits. These criteria have been developed for application to analyses performed with alternative source terms. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting the stated limits demonstrates adequate protection of public health and safety. It is thus concluded that the margin of safety will not be reduced by the implementation of the changes.