VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

July 31, 2001

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

Serial No.	01-037A
NLOS/GDM	R0
Docket Nos.	50-280, 281
License Nos.	DPR-32, 37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
ALTERNATE SOURCE TERM - PROPOSED TECHNICAL SPECIFICATION CHANGE

In a letter dated April 11, 2000 (Serial No. 00-123), Virginia Electric and Power Company (Dominion) submitted a license amendment request for implementation of the Alternate Source Term (AST) as the plant design and licensing bases for Surry Power Station Units 1 and 2. Supplemental responses to NRC requests for additional information were provided on August 28 and November 20, 2000 and April 11, 2001.

In our April 11, 2001 letter (Serial No. 01-037), we noted that extensive calculational work was still required to address questions previously raised by the NRC, and that a future submittal would be necessary to allow sufficient time to complete the work. This effort has now been completed and the appropriate information is provided in the enclosure. Also, a subsequent conference call was held with the NRC staff on May 15, 2001 to address two additional questions that had been provided by the Surry NRC Project Manager, Gordon Edison. At the conclusion of the conference call, Dominion agreed to provide additional information to the NRC to facilitate the staff's continued review of the AST license amendment request. This information is also provided in the enclosure. As previously noted in the April 11, 2001 letter, a revision to the proposed Basis section of Technical Specification 3.10 is required based on the additional analysis work that has been completed. This Basis revision will be provided separately in a later submittal.

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

Very truly yours,

Eugene S. Grecheck
Vice President - Nuclear Support Services

A001

Enclosure

Commitment made in this letter:

1. A revision to the proposed Basis section of Technical Specification 3.10 will be provided in a future submittal.

cc: U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, S.W.
Suite 23 T85
Atlanta, Georgia 30303-8931

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck, who is Vice President - Nuclear Support Services, 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 31st day of July, 2001.

My Commission Expires: 3-31-04.

Maggie McClure

Notary Public

(SEAL)

Enclosure

Response to NRC Request for Additional Information
Alternate Source Term

Surry Power Station Units 1 and 2

Dominion

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
ALTERNATE SOURCE TERM TECHNICAL SPECIFICATION CHANGES
SURRY POWER STATION, UNITS 1 AND 2
SUPPLEMENTAL INFORMATION TO ORIGINAL REPLIES RE: QUESTIONS IN LETTER 01-037 (APRIL 11, 2001)

NRC Question No. 6

The following questions relate to Attachment 3 (Question No. 7) of your November 20, 2000 supplemental alternate source term submittal, that provided details used in the calculation of atmospheric dispersion factors.

a. Determine the wind sector which applies to the number 4 in the first column of figures in the PAVAN input file (denoted as Table 3 and labeled 'PAVAN Input File – Unit 1 Reactor Release Ground-Level Release').

b. Provide the reason for use of the upper wind speed increments indicated in the joint frequency tables generated as PAVAN input.

c. Examine the reason why the apparent output from ARCON96 runs had a time interval of 2-8 hr and these atmospheric dispersion factors were entered in the calculation in the 0-8 hr column. These values are found in the LOCA (Table 5) and Fuel Handling Accident (Table 3) calculations provided as Attachment 1 (Question No. 1) to the November 20, 2000 supplemental alternate source term submittal.

Dominion Supplemental Response

a. Answer provided in previous letter 01-037, dated April 11, 2001.

b. The wind speed bins were incremented in miles per hour and the wind speed data input into these bins were in meters per second. As a result, the relatively high wind speed categories of 12.0-14.9 m/s, 15.0-18.9 m/s and 19.0-23.0 m/s were included in the joint frequency distribution tables. The highest wind speed in the data was in the 9.0-11.9 m/s category. Dominion has re-calculated the atmospheric dispersion factors using wind speed bins consistent with meters per second. Revised joint frequency distribution tables and PAVAN input files are provided in Attachment 2. The revised atmospheric dispersion factors were used in revised LOCA and FHA offsite dose calculations. The results are presented in the accompanying report (Attachment 1).

c. The column heading "0-8 hr" was a typographical error. The column heading should have been "2-8 hr", consistent with ARCON96 output. As a result of this discrepancy, the 0-8 hr atmospheric dispersion factors were used for both the 0-2 hour interval and the 2-8 hr interval. This resulted in non-conservative dose consequences. The Surry LOCA and FHA onsite dose consequences have been re-calculated with the correct ARCON96 atmospheric dispersion factors. The results are presented in the accompanying report (Attachment 1).

NRC Question No. 8

Provide results of a sensitivity calculation that determines the maximum assumed control room unfiltered air inleakage values for a LOCA and a FHA that would result in reaching the control room TEDE limit.

Dominion Supplemental Response

As indicated in the original response, it has been confirmed that the LOCA event has a smaller maximum allowed unfiltered inleakage than that for the FHA event. The LOCA and FHA events have been reanalyzed with revised assumptions and inputs as the result of review questions received from NRC staff. The results of the maximum allowable inleakage sensitivity case for the LOCA event are presented below.

LOCA Analysis Results

Release Pathway	Control Room Dose 10 cfm unfiltered inleakage (rem TEDE)	Control Room Dose 500 cfm unfiltered inleakage (rem TEDE)	EAB Dose (rem TEDE)	LPZ Dose (rem TEDE)
Containment Leakage	0.06	1.00	19.93	1.16
ECCS Leakage	0.48	1.75	2.39	1.10
RWST Backleakage	0.59	2.11	1.69	1.31
Total Dose	1.13	4.86	24.01	3.57
Acceptance Criteria	5.0	5.0	25.0	25.0

ATTACHMENT 1

REVISED ALTERNATIVE SOURCE TERM ANALYSIS REPORT

ASSESSMENT OF ACCIDENT RADIOLOGICAL CONSEQUENCES USING NUREG-1465 METHODOLOGY

SURRY POWER STATION UNITS 1 AND 2

DOMINION

Revision 1 – July 2001

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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). Revision 1 of this report has been prepared to reflect changes in certain analytical assumptions and results that were incorporated in response to review questions received from NRC Staff.

The evaluations documented herein have in general employed the detailed methodology contained in Regulatory Guide 1.183 (23) for use in design basis accident analyses for alternative source terms. Draft Regulatory Guide DG-1081 (3) was employed during preparation of Revision 0 of this report. Certain minor assumption changes were made to reflect the final guidance of RG-1.183. All citations of DG-1081 have been changed to indicate RG-1.183 in Revision 1 of this report. Where alternative approaches to those specified in RG-1.183 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 Dominion 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 (23) 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 RG-1.183, in conjunction with the total effective dose equivalent (TEDE) 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 23, 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 RG-1.183 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)	4.61E-3 sec/m^3
Low Population Zone, LPZ	
0 – 8 hours	2.01E-4 sec/m^3
8 – 24 hours	1.22E-4 sec/m^3
24 – 96 hours	4.18E-5 sec/m^3
96 – 720 hours	8.94E-6 sec/m^3
<u>Breathing Rates</u>	
Control Room	3.5E-4 m^3/sec
Offsite (EAB & LPZ)	
0 – 8 hours	3.5E-4 m^3/sec
8 – 24 hours	1.8E-4 m^3/sec
24 – 720 hours	2.3E-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

<u>Key Operator Actions</u>
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, since hatch closure requires actions from inside containment. 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 RG-1.183, 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 containment ventilation purge 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 RG-1.183 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 Dominion Generation 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. Minor roundoff in certain criteria as issued in final Regulatory Guide 1.183 is reflected in the table.

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	6.3 rem TEDE
Fuel Handling Accident	5 rem TEDE	6.3 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 RG-1.183 (23). The LOCA analysis results have been revised to reflect: 1) changes in offsite X/Q modeling assumptions that were incorporated in response to review questions received from NRC Staff, 2) a change in assumed control room air intake point made possible by a plant design change and 3) breathing rates as stipulated by RG-1.183. These revisions affected the assumed X/Q values that are indicated below in Sections 3.1.3.1 and 3.1.3.2. RWST flowpath modeling was also revised to reflect a plant design change and reduced safety injection system check valve leakage. These changes affected the RWST release flow rates and the ECCS backleakage flowrates to the RWST.

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 RG-1.183. 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 References (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
Sb-129	2.13E+07	6.44E+02	2.64E-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)
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
Ce-143	1.02E+08	3.39E+03	4.77E-02
Ce-144	9.10E+07	3.74E+05	3.16E-03

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)
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
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

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)
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. For the Main Control Room, X/Qs were calculated for these source points: Unit 1 and Unit 2 Containment building, Ventilation Vent No. 2, and East and West louvers of the Auxiliary Building. 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). In response to a question from NRC Staff, the X/Q values selected for use in the analysis have been modified. This item was discussed in the written response provided in Reference (24). The change involved correction of a discrepancy in reporting calculated X/Q results for the 0-8 hour interval. This discrepancy contributed to inadvertently using the 0-8 hour X/Q values for both the 0-2 hour and 2-8 hour intervals in the original analysis. Since the calculated X/Q for the 0-2 hour interval is greater, this resulted in non-conservative dose consequences. Also, new source-to-receptor pairs were selected to reflect completion of portions of Design Change 99-109, which terminates operation of Turbine Building non-safety related fans upon automatic or manual isolation of the control room. Securing the Turbine Building supply fans closes the louvers that were previously used as receptor points for control room dose calculations, allowing the selection of less limiting receptors. This change involved no revised X/Q calculations.

The ARCON96 algorithm generates X/Q values which are mildly dependent upon the vertical velocity of the effluent stream. This implies that a lower vertical velocity creates less dispersion and a larger X/Q. Therefore, the vertical velocities assumed in ARCON96 for the Ventilation Vent No. 2 release path are a lower bound of the potential system configurations, with and without offsite power available. The revised values used for the main control room X/Qs are reported 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. For the EAB, X/Qs were calculated for these source points: Unit 1 and Unit 2 Containment building, Ventilation Vent No. 2, and East and West louvers of the Auxiliary Building. The EAB modeling was simplified by utilizing the current Technical Specifications (TS 5.1) definition of the EAB such that the onsite release points were all modeled at the same minimum distance (1650 ft). Previous analyses assumed the EAB is a circle of 1650 ft radius, centered on the Unit 1 Containment building (per definition in UFSAR Section 2.1.2). For the LPZ, the same 5 source points were modeled. For both the EAB and LPZ, the most conservative X/Q among those calculated was assumed in the radiological analyses.

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). In response to a question from NRC Staff, the offsite X/Q values used in the analysis have been modified. This item was discussed in the written responses provided in Reference (24). The change involved correction of an inconsistency in wind speed unit translation (miles/hour to meters/second) between the wind speed inherent in the meteorological data and the wind speed bin structure assumed in preparing the joint frequency distribution tables for use in the PAVAN code. Use of the revised wind speed increments resulted in more conservative

X/Q values. The 0-2 hr X/Q value was used in the calculation of the worst 2 hour EAB dose. The revised values used for the offsite X/Qs are reported 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, resulting in the analysis assuming that 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 RG-1.183, 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 aerosol removal coefficients 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

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.

3.1.5.1 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

3.1.5.2 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 200 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. This leakage flowrate has been reduced to reflect current system performance as validated by inservice inspections. The revised flowrate is indicated on Table 3.1-7.

The pathway for release from the tank has been altered by a design change that was implemented after issuing Revision 0 of this report. The modification involved uncapping a previously sealed gooseneck pipe at the top of the tank, allowing communication to the atmosphere. The top of the RWST contains a separate 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 in the following fashion. It is assumed that the Auxiliary Ventilation system draws outside air into the RWST at a rate of 1000 cfm and that 1000 cfm of air (containing iodine assumed to evolve within the tank) is displaced through the vent pipe into the Safeguards Building. This air is then discharged through Ventilation Vent No. 2 to the atmosphere. This assumed flowrate through the RWST bounds the post-modification tested flowrate for operation with the safety-related Auxiliary Ventilation system fans. Holdup in the RWST is modeled, based on the free tank volume, but creates minimal benefit because of the relatively high release flow rate. 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.

3.1.5.3 Auxiliary Ventilation System Model

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 Area. This degree of dependence upon the filtration was previously assumed in order to maintain the current licensing basis of not including the leakage from a passive failure (e.g., pump seal). It is no longer necessary to accommodate the effects of a passive failure in radiological analyses, per the guidance in Appendix A of RG-1.183. Implementation of the AST allows the Surry licensing basis to be revised such that an ECCS passive failure is not longer postulated, and its direct or indirect effects need not be considered. The Technical Specifications LCOs for operability of the auxiliary ventilation safety-related filters are maintained for initial implementation of the AST, but may be considered for future deletion.

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)	200 cc/min
Effluent Flowrate from RWST to Atmosphere	1000 cfm
RWST Free Volume	53,350 ft^3

MCR Atmospheric Dispersion Factors	Containment	ECCS Leakage
0 – 2 hours	6.74E-4 sec/m^3	6.95E-4 sec/m^3
2 – 8 hours	5.18E-4 sec/m^3	5.40E-4 sec/m^3
8 – 24 hours	2.22E-4 sec/m^3	2.30E-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
Realign Auxiliary Ventilation to Safeguards Area Exhaust	Prior to RMT switchover

3.1.6 LOCA Analysis Results

The results of the LOCA dose analysis are presented in Table 3.1-8, and have been revised to reflect the changes in assumptions that were incorporated in response to review questions received from NRC Staff. 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 RG-1.183. 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	0.06	19.93	1.16
ECCS Leakage	0.48	2.39	1.10
RWST Backleakage	0.59	1.69	1.31
Total Dose	1.13	24.01	3.57
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 RG-1.183 (23), although some significant exceptions are proposed, with accompanying justification. The Fuel Handling Accident analysis results have been revised to reflect changes in the fuel rod fission gas inventory assumptions that were incorporated in response to review questions received from NRC Staff. These revisions affect the assumed total inventory available for release from the failed rods and are indicated below in Section 3.2.2. Several other changes were made as follows:

1) Both the onsite and offsite X/Q values were revised consistent with the discussion of LOCA X/Q values

2) Assumed Iodine speciation: 99.85% elemental 0.15% organic in accordance with RG-1.183

3) Breathing rates as indicated in RG-1.183.

4) Offsite dose acceptance criterion of 6.3 rem TEDE in accordance with RG-1.183.

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

3.2.2.1 General Considerations

Reference (23), in Section 3.2, 'Release Fractions,' provides Table 3, 'Non-LOCA Fraction of Fission Product Inventory in Gap.' The NRC Staff guidance in Appendix B of Reference (23) refers to the Table 3 values as acceptable values for use in FHA analyses. The detailed discussion that follows describes a rationale for use of an alternative to the Reference (23) 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. The first two of these key elements listed above are described below as applied in the FHA event analysis. The third element (release mechanism) is treated merely by assumption that the rods have failed; no specific mechanism is postulated. 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.

3.2.2.2 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.

Individual assemblies in each batch can be operated at powers that exceed the batch average power modeled in the ORIGEN2 runs. It is therefore necessary to increase the curie inventory by applying a peaking factor adjustment. Section C.3.1 of RG-1.183 indicates that radial peaking factors from the Core Operating Limits Report (COLR) should be applied in determining the inventory of the damaged rods. The maximum allowed radial peaking factor from the Surry COLR, including uncertainties, is 1.62 (for a peak rod). This assumed peaking factor is applied to increase the calculated total inventory of the once and twice burned assemblies. A reduced value of 1.188 is applied to the thrice-burned assemblies. This adjustment is conservative, since its rationale is to obtain the inventory of the damaged rods (for Surry, all of the rods in a single assembly). It would also be consistent with RG-1.183 to apply an adjustment based upon assembly average powers.

Table 3.2-1 presents results of the ORIGEN2 calculations, in terms of total inventory for a single assembly from each region modeled. The isotopes included in Table 3.2-1 are a combination of fission products and the daughter products of decay from other isotopes such as Tellurium. The results of this calculation demonstrate that a fuel assembly at the end of its first cycle of irradiation contains the maximum total inventory of radiologically significant isotopes (primarily iodine). To ascertain the most limiting fuel assembly inventory actually available for release, it is necessary to adjust (i.e., multiply) the calculated total inventory by the fraction that is released into the fuel rod/cladding gap. This adjustment is made for each of the three assemblies modeled, since releasable inventory is the product of total inventory and gap fraction and it is not known a priori for which assembly this quantity is maximized. A detailed discussion concerning determination of the specific fraction of this total inventory that is actually in the fuel rod/cladding gap and thus available for release is discussed in Section 3.2.2.4 below.

Table 3.2-1

Single Fuel Assembly Total Inventory by Isotope For Each Fuel Region Modeled (after 100 Hours of Decay)

Isotope	Once Burned Assembly Activity (Curies)	Twice Burned Assembly Activity (Curies)	Thrice Burned Assembly Activity (Curies)
I-130	5.721E+01	1.196E+02	1.155E+02
I-131	5.006E+05	4.757E+05	3.805E+05
I-132	4.228E+05	3.963E+05	3.160E+05
I-133	5.301E+04	4.831E+04	3.821E+04
I-135	3.786E+01	3.481E+01	2.759E+01
Kr-85	4.468E+03	8.023E+03	2.051E+04
Kr-88	1.384E-05	1.003E-05	7.443E-06
Kr-83m	1.093E-07	8.661E-08	6.615E-08
Kr-85m	3.924E-02	2.935E-02	2.203E-02
Xe-133	1.002E+06	9.163E+05	7.280E+05
Xe-135	1.824E+03	1.682E+03	1.457E+03
Xe-131m	7.509E+03	7.151E+03	5.727E+03
Xe-133m	1.848E+04	1.706E+04	1.354E+04
Xe-135m	6.066E+00	5.578E+00	4.419E+00

3.2.2.3 Fraction of Available Inventory Existing In a Releasable Form

Section 3.2, Table 3 of Regulatory Guide 1.183 (23) 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	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

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, Dominion considers it 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 events 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. Other fuel characteristics of rods predicted to fail, such as power level and burnup, will also affect the fission gas release. 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 gap fraction values is that Category 1 events have the smallest fraction, followed by Category 2, then Category 3 with the maximum value. For the present application, only values associated with the FHA event are proposed.

3.2.2.4 Determination of Fission Gas Gap Fraction

For the FHA event, Dominion proposes use of an alternative to the fission gas release gap fractions recommended in RG-1.183. This approach involves use of available industry data to justify a fission gas gap fraction that is deemed more applicable to the FHA event analysis than the values in RG-1.183. These data were discussed with NRC Staff in a February 2000 meeting with the NEI Source Term Task Force and were included in the comments of Reference (25). The Reference (25) proposal involved an envelope, in the form of a piece-wise linear function of burnup, that was constituted to provide margin with respect to the underlying data. Figure 3.2-1 presents the proposed limit envelope and data as contained in Enclosure 2, Appendix A of

Reference (25). The data are a composite of the measured points presented in References (26) and (27).

Dominion has evaluated the proposed limit curve from Figure 3.2-1 to ascertain whether it provides an appropriate bound for fission gas release behavior of LWR fuel operating under Surry fuel management practices. NRC Staff have indicated that the proposed envelope from Reference (25) was considered during development of RG-1.183. For this assessment, the evaluation has focused upon whether the margin between the proposed envelope and the underlying data can accommodate the following considerations:

1) rod powers associated with operation of long-lived high power density cores
2) effects of Condition I operational power transients

Reference (25) provided little characterization of the underlying data, and no assessment regarding either of the two issues above. Both of these considerations have some potential to impact either overall fission gas release or fission gas migration to the fuel/cladding gap. The basic question is whether the data, after considering the potential impact of these issues, remains bounded by the envelope proposed in Figure 3.2-1. It has been concluded that the proposed envelope does not bound the expected fission gas release for Surry, after considering the Surry core management and rod power conditions. However, accompanying details in Reference (26) allowed the definition of a method for determining fission gas gap fractions applicable to Surry rods for the FHA analysis. This method is described next.

The data in Reference (26) present fission gas release as a function of time-average linear heat generation rate (LHGR), for several different pellet geometries and different values of as-fabricated open porosity. The data show a strong trend of increasing fission gas release (i.e., gap fraction) as time-average LHGR increases.

Reference (27) includes a discussion of fission gas release mechanisms that are dominant for low temperature (i.e., normal operation) irradiation. Since fuel temperature is relatively low by the time that significant quantities of fission gas are generated, the contribution from diffusion effects is small. Reference (27) cites knockout and recoil as the mechanisms that dominate for such low temperature operation. In addition, Reference (27) notes that a correlation exists between the open porosity of the fuel and the resultant fission gas release. Reference (26) reports measurements of fission gas release that supports such a relationship. Three different pellet designs are reported in Reference (26), each with a different initial open porosity. The open porosity values range from 0.14% to 0.48%, with the reported fission gas release increasing as open porosity increases. The reported data are very well represented by a linear regression for each different value of porosity, creating a family of curves for fission gas release versus LHGR (one curve for each value of porosity).

Core design predictions were surveyed for four recent Surry cycles to calculate the time-average LHGR for typical fuel assemblies discharged after 3 cycles of irradiation. These calculated values were based upon assembly average rod powers, to maintain consistency with the assumption that the FHA event releases the gap inventory for all the rods in a single fuel assembly. This assumption accounts for the fact that gap inventory postulated to be released in the FHA event is the aggregate inventory of all 204 fuel rods in one assembly, mixed in the water surrounding the assembly. The calculated Surry LHGR values were used to estimate the fission gas release fraction at the end of 3 cycles of irradiation, based upon a linear regression of the Reference (26) data for pellet open porosity of 0.48%. Data for these rods most closely matches the open porosity of the Surry fuel rods, which is approximately 0.45%. This calculation in effect involved an extrapolation, since the Surry time-average LHGR values were greater than those for the data points. Refer to this intermediate result as the Data-Extrapolated Gap Fraction.

The Data-Extrapolated Gap Fraction represents the value that would be expected for Surry fuel, under two conditions: 1) if it were irradiated to the final burnup of the underlying data points (60 GWD/MTU) and 2) if this irradiation occurred while maintaining the Surry time-average LHGR as indicated on Table 3.2-2. This calculated gap fraction is overly conservative, since the additional irradiation to 60 GWD/MTU would inherently decrease the time-average LHGR. To obtain a more appropriate estimate of the 3-cycle Surry gap fraction, the actual Surry burnup is used to interpolate from the Data-Extrapolated Gap Fraction (applicable to 60 GWD/MTU) to a result for the actual Surry burnup. This interpolation employs the assumption that fission gas gap fraction has a linear relationship to burnup, but only when benchmarked to data that is applicable for the expected final rod burnup. Once the 3-cycle fission gas release fraction was established, the values corresponding to the end of first and second cycle irradiation were conservatively estimated by assuming a linear interpolation (versus burnup) toward a defined initial point of zero gap fraction at zero burnup. References (26) and (27) conclude that gap fraction burnup dependency is minimal at low burnups, but is much greater at high burnup. These conclusions support the conservatism of assuming a linear relationship of gap fraction with burnup. Table 3.2-2 summarizes the results of the rod power calculations and the estimated Surry gap fraction values for each of the four cycles studied, applying the steps described above. To include some margin for future core design flexibility, the values assumed in the FHA radiological analysis are increased from those presented in Table 3.2-2. The gap fraction values assumed in the FHA radiological analysis are summarized below.

Once-Burned Gap Fraction = 3.0%
Twice-Burned Gap Fraction = 5.35%
Thrice-Burned Gap Fraction = 6.0%

Dominion will ensure that future Surry core designs continue to be bounded by the key parameter values used in determining these fission gas gap fraction results.

Each of these values is applied to the total inventory of I-131 for the respective assembly to determine the assembly that has the maximum releasable inventory for use in the FHA radiological analysis. The RG-1.183, Table 3 gap fraction values are assumed for the other 13 isotopes listed in Section 3.2.2.2. The steps involved in determining the limiting assembly releasable inventory are described in Section 3.2.2.6.

Table 3.2-2

Surry Assembly Average Rod Fission Gas Gap Fractions (for FHA Analysis) [1]

Fuel Cycle	Assembly Irradiation	Maximum BU (GWD/MTU)	Time-Avg LHGR (kw/ft)	Fission Gas Release (%)
S1C16	Once-Burned	22.343	N/A	2.23
	Twice-Burned	42.925	N/A	4.29
	Thrice-Burned	48.453	7.11	4.85
S1C17	Once-Burned	23.033	N/A	2.24
	Twice-Burned	45.263	N/A	4.41
	Thrice-Burned	50.285	7.05	4.89
S2C16	Once-Burned	21.713	N/A	2.38
	Twice-Burned	44.338	N/A	4.86
	Thrice-Burned	49.148	7.35	5.39
S2C17	Once-Burned	23.271	N/A	2.48
	Twice-Burned	44.042	N/A	4.70
	Thrice-Burned	51.475	7.28	5.49

[1] Values based upon total core power of 2546 MWt and core average LHGR of 6.45 kw/ft

3.2.2.5 Effects of Condition I Operational Transients

In addition to the time-average rod power history, the transient power changes that rods experience also have the potential to impact fission gas release. This is primarily a concern only for extreme power ramps, such as those reported in Reference (27). In fact, Reference (27) concluded that normal variations in power typical of the irradiation history of power operation do not markedly affect fission gas release as measured at end of life. This effect is inherent in the reported data, which involved normal operational transients and power operation.

The fission gas measurement data included in EPRI Report TR-103302-V2 indicate that fission gas release increases as rod power increases. However, the mechanisms believed to be active for rods in the burnup range evaluated are not extremely sensitive to small changes in rod power. The Reference (26) data were presented in terms of time-average rod linear heat generation rate (LHGR). The approach proposed for Surry employs power history design inputs based upon core design calculations that assume full power steady-state operation within the COLR peaking factor limits. This represents an effective upper bound on time-averaged LHGR, since plant operational controls and core design calculations ensure that the COLR peaking factor limits will be met. Existence of such an upper limit, indicates that the effect of operational transients will be to decrease the time-averaged LHGR, resulting in less overall fission gas release than is being modeled. This is due to the fact that since the time-average LHGR values are based upon full power steady-state operation, operational transients will result in net power reductions with respect to this baseline. Based on these considerations, it is concluded that the assumed fission gas gap fraction values accommodate the effect of expected Condition I transients, with no further adjustment.

3.2.2.6 Determination of Limiting Assembly and Activity Released

Using the fission gas gap fractions determined above, the fuel assembly with the maximum releasable inventory can be determined. This assembly will yield the limiting dose for the FHA event. The steps involved in the proposed methodology (described in detail in Sections 3.2.2.2 through 3.2.2.4) are summarized below.

1. Perform ORIGEN2 calculations of batch average isotopic inventory assuming representative average power distributions for the regions modeled (once, twice, thrice-burned).

2. Adjust ORIGEN2 inventory of each assembly to reflect the maximum assembly power that bounds those achievable from core management plans. (Table 3.2-1 reflects this result)

3. Multiply total isotopic inventory (from Table 3.2-1) for each assembly by the appropriate gap fractions. For I-131, the gap fractions from Section 3.2.2.4 are used. For all other isotopes, apply the RG-1.183 gap fractions. This yields the releasable inventory for each assembly modeled.

4. Select assembly with maximum releasable inventory for FHA analysis. Table 3.2-3 indicates that the twice-burned fuel assembly has maximum I-131 gap inventory and thus is limiting for FHA dose.

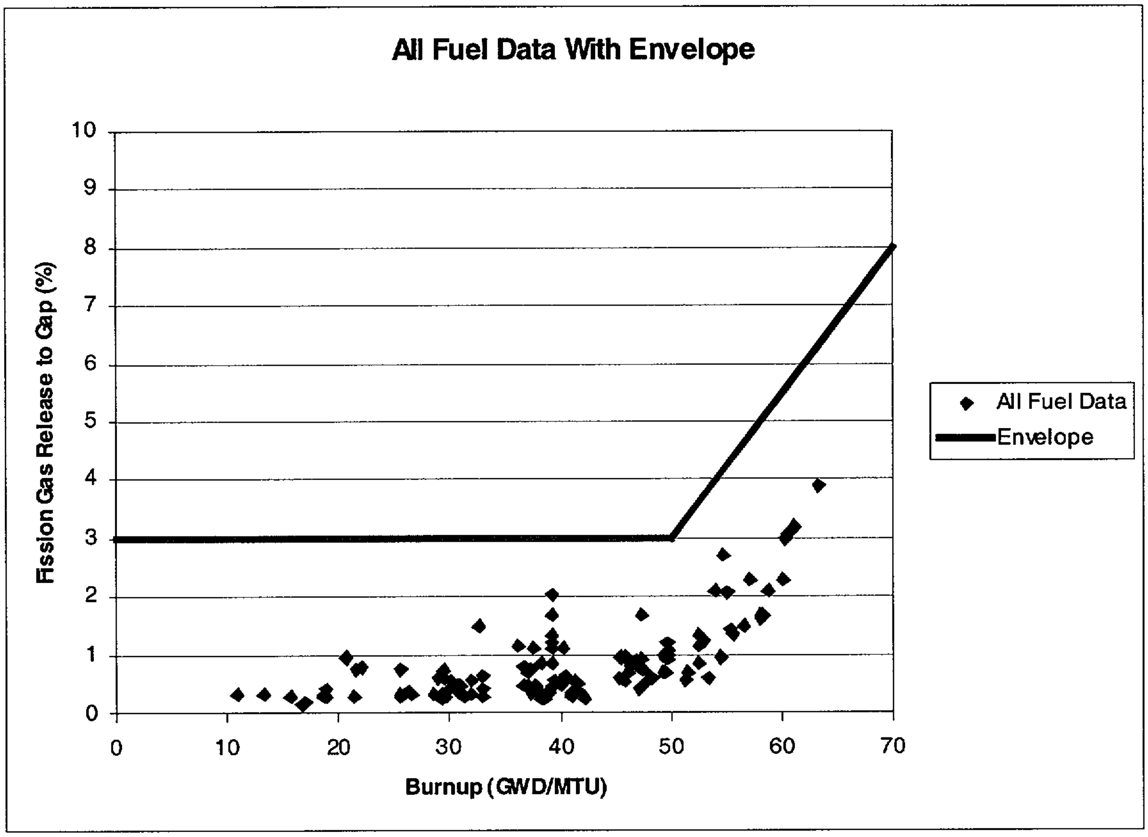
Table 3.2-3 summarizes the results for total assumed activity in the fuel rod gap for each of the assemblies modeled, by isotope analyzed in the FHA event. This activity is the number of curies available for release from the failure of the cladding from all the rods in one fuel assembly, determined in the manner described above. This represents the assumed activity that is available for release to the water surrounding the failed fuel assembly.

Table 3.2-3 – Single Fuel Assembly Gap Inventory by Isotope for Each Region Modeled (after 100 Hours of Decay)

Isotope	Once Burned Assembly Activity (Curies)	Twice Burned Assembly Activity (Curies)	Thrice Burned Assembly Activity (Curies)
I-130	2.861E+00	5.981E+00	5.774E+00
I-131	1.502E+04	2.545E+04	2.283E+04
I-132	2.114E+04	1.982E+04	1.580E+04
I-133	2.651E+03	2.416E+03	1.911E+03
I-135	1.893E+00	1.741E+00	1.379E+00
Kr-85	4.468E+02	8.023E+02	2.051E+03
Kr-88	6.918E-07	5.013E-07	3.722E-07
Kr-83m	5.464E-09	4.331E-09	3.308E-09
Kr-85m	1.962E-03	1.467E-03	1.102E-03
Xe-133	5.012E+04	4.582E+04	3.640E+04
Xe-135	9.121E+01	8.409E+01	7.285E+01
Xe-131m	3.754E+02	3.575E+02	2.863E+02
Xe-133m	9.242E+02	8.529E+02	6.772E+02
Xe-135m	3.033E-01	2.789E-01	2.210E-01

Figure 3.2-1

Fission Gas Release Data and Industry-Proposed Envelope for FHA Analysis



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

The source points considered were the Unit 1 and Unit 2 Containment buildings, Ventilation Vent No. 2, and the Auxiliary Building East and West louvers. 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 intakes, the turbine building rollup doors, the EAB and the LPZ. The Turbine Building fresh air louvers are no longer modeled as receptor points due to the completion of portions of DCP-99-109, which terminated power to non-safety Turbine Building fans upon occurrence of control room isolation (manual or automatic). In response to a question from NRC Staff, the X/Q values selected for use in the analysis have been modified. This item was discussed in the written response provided in Reference (24). The change involved correction of a discrepancy in reporting calculated control room X/Q results for the 0-8 hour interval. This discrepancy contributed to inadvertently using the 0-8 hour X/Q values for both the 0-2 hour and 2-8 hour intervals in the original analysis. Since the calculated X/Q for the 0-2 hour interval is greater, this resulted in non-conservative dose consequences. The revised values used for the onsite (Main Control Room) atmospheric dispersion factors in the FHA analysis are reported in Table 3.2-4.

3.2.4 Determination of Offsite Atmospheric Dispersion Factors (X/Q)

The offsite atmospheric dispersion factors (EAB and LPZ) used for the FHA analysis are the same as those used for LOCA. They are reported in Table 1.3-1.

3.2.5 FHA Analysis Assumptions & Key Parameter Values

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.

3.2.5.1 Effluent Flowrates Assumed for FHA Release Paths

In Appendix B of RG-1.183 (23), 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 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 3.2-4.

3.2.5.1 X/Q Selection for Multiple Containment Release Paths

The X/Q values used to model various release pathways are presented in Table 3.2-4. 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 3.2-4 summarizes analysis assumptions and key input parameter values that are unique to the FHA analysis cases.

Table 3.2-4
Analysis Assumptions & Key Parameter Values
Employed Only in Fuel Handling Accident Analysis

<u>Containment Parameters</u>	
Release Flowrate (0 – 720 hours)	2,000 – 36,000 cfm [1]
Free Volume (for holdup; 50% of total)	9.315E5 ft^3
<u>Core and Fuel Assembly Characteristics</u>	
Number of Fuel Assemblies in Core	157
Maximum Fuel Assembly Radial Peaking Factor	1.62, 1.62, 1.188 [2]
Assumed Iodine Physical Form In Gap	99.85% elemental 0.15% organic

<u>MCR Atmospheric Dispersion Factors</u>

	Equipment Hatch	Personnel Airlock	Fuel Building/Purge
0 –2 hour	6.74E-4 sec/m^3	1.07E-3 sec/m^3	6.97E-4 sec/m^3
2 – 8 hour	5.18E-4 sec/m^3	9.03E-4 sec/m^3	5.43E-4 sec/m^3
8 – 24 hours	2.22E-4 sec/m^3	3.87E-4 sec/m^3	2.31E-4 sec/m^3
24 – 96 hours	1.66E-4 sec/m^3	2.73E-4 sec/m^3	1.71E-4 sec/m^3
96 – 720 hours	1.20E-4 sec/m^3	1.87E-4 sec/m^3	1.22E-4 sec/m^3

<u>Offsite Atmospheric Dispersion Factors (EAB)</u>

2 – 8 hour	2.60E-3 sec/m^3
8 – 24 hour	1.96E-3 sec/m^3
24 – 96 hour	1.05E-3 sec/m^3
96 – 720 hour	4.32E-4 sec/m^3

<u>Miscellaneous</u>	
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)	3,500 - 80,000 cfm[1]
<u>Key Operator Actions</u>	<u>Timing of Action</u>
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.
[2] Values are for once, twice and thrice-burned assemblies

3.2.6 FHA Analysis Results

The results of the FHA dose analysis are presented in Table 3.2-5, and have been revised to reflect the changes in assumptions that were incorporated in response to review questions received from NRC Staff. 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 RG-1.183. It can be observed from the table that each of the results meets the dose acceptance criteria.

Table 3.2-5 – 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)	0.16	6.27	0.27
Acceptance Criteria	5.0	6.3	6.3

[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., RG-1.183) 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 RG-1.183 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 RG-1.183.

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.

7. Letter, L. A. Walsh (Westinghouse Owners' Group Steam Generator Tube Uncovery Task Team) to R. C. Jones (USNRC), "Westinghouse Owners' Group Steam Generator Tube Uncovery Issue," OG-92-25, March 31, 1992.

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.

6.0 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. [DELETED]

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.

23. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," USNRC, Office of Nuclear Regulatory Research, July 2000.

24. Letter, David A. Christian to USNRC, "Surry Power Station Units 1 and 2-Request for Additional Information-Alternate Source Term-Proposed Technical Specification Change," Serial No. 01-037, April 11, 2001.

6.0 References (continued)

25. Letter, David J. Modeen (NEI) to Ms. Annette Vietti-Cook (Secretary to USNRC), Forwarding Comments on Draft Regulatory Guide DG-1081, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,* March 31, 2000.

26. "Hot Cell Examination of Extended Burnup Fuel from Calvert Cliffs-1," EPRI, Palo Alto, CA,: July 1994, EPRI TR-103302-V2.

27. S. R. Pati and A. M. Garde, "Fission Gas Release From PWR Fuel Rods at Extended Burnups," Proc. ANS Topical Meeting on Light Water Reactor Fuel Performance, Orlando, Florida, April 21-24, 1985.

ATTACHMENT 2

(QUESTION NO. 6)

REVISED JOINT FREQUENCY DISTRIBUTION TABLES AND PAVAN INPUT FILES

TABLE 1

JOINT FREQUENCY DISTRIBUTION TABLES FOR 1982-1986 DATA WITH STANDARD M/S BINS

1 1982-86 SURRY MET DATA 10:56 WEDNESDAY,
FEBRUARY 28, 2001 1

STABILITY CLASS A

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	1	1	0	2	0	0	0	0	12
B) 0.51-.75	1	0	1	1	1	0	4	2	17
C) 0.76-1.0	1	1	0	7	5	6	4	3	49
D) 1.1-1.50	24	15	16	55	60	50	66	37	496
E) 1.51-2.0	69	41	57	171	171	173	173	72	1520
F) 2.1-3.00	186	174	186	366	283	238	354	77	3557
G) 3.1-5.00	293	232	181	289	201	12	45	15	3283
H) 5.1-7.00	38	3	6	2	4	0	0	1	360
I) 7.1-10.0	0	0	0	2	0	0	0	0	31
TOTAL	613	467	447	895	725	479	646	207	9325

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	1	0	0	0	1	2	3	1	12
B) 0.51-.75	1	0	1	3	1	0	1	0	17
C) 0.76-1.0	5	7	2	0	1	1	2	4	49
D) 1.1-1.50	22	22	21	12	24	23	28	21	496
E) 1.51-2.0	44	55	49	55	97	128	94	71	1520
F) 2.1-3.00	86	194	191	162	244	359	251	206	3557
G) 3.1-5.00	47	290	437	220	137	323	274	287	3283
H) 5.1-7.00	3	42	81	21	27	45	50	37	360
I) 7.1-10.0	0	1	3	2	11	5	6	1	31
TOTAL	209	611	785	475	543	886	709	628	9325

STABILITY CLASS B

TABLE OF WINSPEED BY SECTOR

WINSPEED FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
B) 0.51-.75	0	1	0	1	0	0	1	0	3
C) 0.76-1.0	0	2	0	0	1	2	5	0	22
D) 1.1-1.50	11	2	3	10	11	12	19	9	144
E) 1.51-2.0	16	8	9	25	18	23	23	7	254
F) 2.1-3.00	37	31	26	34	39	10	16	8	422
G) 3.1-5.00	58	37	28	27	18	2	5	6	395
H) 5.1-7.00	3	0	3	1	1	0	0	0	51
I) 7.1-10.0	0	0	0	2	0	0	0	0	6
TOTAL	125	81	69	100	88	49	69	30	1297

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
B) 0.51-.75	0	0	0	0	0	0	0	0	3
C) 0.76-1.0	1	0	3	2	3	1	2	0	22
D) 1.1-1.50	13	7	9	10	6	13	4	5	144
E) 1.51-2.0	9	15	15	13	21	18	20	14	254
F) 2.1-3.00	7	20	30	35	35	40	27	27	422
G) 3.1-5.00	12	34	49	34	17	27	20	21	395
H) 5.1-7.00	0	9	8	6	7	2	8	3	51
I) 7.1-10.0	0	1	1	0	0	2	0	0	6
TOTAL	42	86	115	100	89	103	81	70	1297

STABILITY CLASS C

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	0	0	0	0	1	0	0	0	1
B) 0.51-.75	0	0	1	0	1	0	3	0	7
C) 0.76-1.0	2	1	0	3	3	1	4	3	27
D) 1.1-1.50	9	7	3	7	14	24	23	13	166
E) 1.51-2.0	16	14	12	17	24	13	13	9	228
F) 2.1-3.00	51	32	37	33	42	21	11	5	455
G) 3.1-5.00	45	40	35	29	16	1	1	6	429
H) 5.1-7.00	3	1	0	2	1	0	0	0	54
I) 7.1-10.0	1	0	0	0	0	0	0	0	5
TOTAL	127	95	88	91	102	60	55	36	1372

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	0	0	0	0	0	0	0	0	1
B) 0.51-.75	1	1	0	0	0	0	0	0	7
C) 0.76-1.0	1	2	4	1	0	1	1	0	27
D) 1.1-1.50	8	14	6	8	9	10	6	5	166
E) 1.51-2.0	8	9	20	12	15	15	18	13	228
F) 2.1-3.00	9	31	41	31	25	22	34	30	455
G) 3.1-5.00	13	40	59	39	26	23	19	37	429
H) 5.1-7.00	1	5	11	2	7	2	8	11	54
I) 7.1-10.0	0	1	0	0	1	0	1	1	5
TOTAL	41	103	141	93	83	73	87	97	1372

STABILITY CLASS D

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	3	1	0	1	3	1	1	4	23
B) 0.51-.75	2	1	0	7	3	5	11	7	72
C) 0.76-1.0	12	17	13	12	18	43	43	29	334
D) 1.1-1.50	54	44	61	52	88	112	79	70	958
E) 1.51-2.0	97	88	106	86	128	130	52	54	1309
F) 2.1-3.00	277	261	293	228	252	93	86	45	2704
G) 3.1-5.00	412	330	283	281	170	22	30	30	2854
H) 5.1-7.00	75	36	26	54	12	6	0	4	561
I) 7.1-10.0	6	1	7	17	3	1	0	0	60
TOTAL	938	779	789	738	677	413	302	243	8875

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	4	0	2	1	1	0	0	1	23
B) 0.51-.75	6	5	5	5	3	5	5	2	72
C) 0.76-1.0	20	28	18	20	17	12	17	15	334
D) 1.1-1.50	48	62	52	52	50	46	35	53	958
E) 1.51-2.0	45	55	75	61	73	79	72	108	1309
F) 2.1-3.00	72	167	160	146	118	114	139	253	2704
G) 3.1-5.00	64	205	208	125	115	107	147	325	2854
H) 5.1-7.00	19	47	40	22	27	32	91	70	561
I) 7.1-10.0	1	7	1	1	6	4	4	1	60
TOTAL	279	576	561	433	410	399	510	828	8875

STABILITY CLASS E

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	3	2	1	5	8	7	8	14	88
B) 0.51-.75	8	5	10	7	6	17	28	29	191
C) 0.76-1.0	21	14	16	18	37	62	97	94	622
D) 1.1-1.50	73	92	77	55	111	290	141	160	1703
E) 1.51-2.0	148	130	151	103	238	294	85	136	2239
F) 2.1-3.00	417	265	243	122	186	117	95	160	4065
G) 3.1-5.00	224	123	115	50	60	16	46	43	2705
H) 5.1-7.00	19	1	9	9	9	5	1	3	228
I) 7.1-10.0	10	3	5	3	7	0	0	0	41
TOTAL	923	635	627	372	662	808	501	639	11882

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	11	10	7	2	3	0	2	5	88
B) 0.51-.75	22	18	8	6	10	8	5	4	191
C) 0.76-1.0	70	47	29	31	23	25	18	20	622
D) 1.1-1.50	125	105	113	110	79	60	51	61	1703
E) 1.51-2.0	181	124	142	137	102	89	67	112	2239
F) 2.1-3.00	354	596	411	294	186	131	178	310	4065
G) 3.1-5.00	201	592	327	187	147	111	187	276	2705
H) 5.1-7.00	12	33	30	18	20	19	26	14	228
I) 7.1-10.0	0	1	4	1	6	0	1	0	41
TOTAL	976	1526	1071	786	576	443	535	802	11882

STABILITY CLASS F

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	4	5	7	3	8	12	28	19	152
B) 0.51-.75	7	5	6	8	18	23	44	49	256
C) 0.76-1.0	22	12	16	27	19	69	117	138	749
D) 1.1-1.50	111	64	36	36	41	107	110	198	1522
E) 1.51-2.0	79	31	15	14	19	11	23	77	1165
F) 2.1-3.00	27	12	3	4	1	1	1	25	865
G) 3.1-5.00	8	1	0	0	0	0	0	2	65
H) 5.1-7.00	1	0	0	0	0	0	0	1	3
I) 7.1-10.0	0	0	0	0	0	0	0	0	1
TOTAL	259	130	83	92	106	223	323	509	4778

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	21	12	11	7	3	3	4	5	152
B) 0.51-.75	25	21	12	5	8	6	6	13	256
C) 0.76-1.0	91	63	40	31	24	23	23	34	749
D) 1.1-1.50	199	117	103	82	85	64	69	100	1522
E) 1.51-2.0	197	191	167	106	82	54	44	55	1165
F) 2.1-3.00	135	278	163	91	63	27	21	13	865
G) 3.1-5.00	7	21	15	2	4	1	0	4	65
H) 5.1-7.00	0	0	0	0	1	0	0	0	3
I) 7.1-10.0	0	0	0	0	1	0	0	0	1
TOTAL	675	703	511	324	271	178	167	224	4778

STABILITY CLASS G

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	15	7	12	10	16	16	30	62	459
B) 0.51-.75	19	15	7	9	24	27	79	99	693
C) 0.76-1.0	30	23	29	28	37	64	119	204	1541
D) 1.1-1.50	39	22	11	15	7	13	65	188	1290
E) 1.51-2.0	7	3	0	1	2	1	5	21	427
F) 2.1-3.00	1	0	0	1	0	1	0	0	170
G) 3.1-5.00	0	0	0	0	0	0	0	0	7
H) 5.1-7.00	0	0	0	0	0	0	0	0	2
TOTAL	111	70	59	64	86	122	298	574	4589

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	73	59	50	31	25	17	21	15	459
B) 0.51-.75	116	81	61	48	37	28	20	23	693
C) 0.76-1.0	278	256	138	98	66	69	56	46	1541
D) 1.1-1.50	245	287	121	70	68	51	45	43	1290
E) 1.51-2.0	104	141	70	19	14	12	13	14	427
F) 2.1-3.00	44	78	28	3	3	3	2	6	170
G) 3.1-5.00	0	0	2	0	3	1	0	1	7
H) 5.1-7.00	0	0	0	1	0	0	0	1	2
TOTAL	860	902	470	270	216	181	157	149	4589

TABLE 2
JOINT FREQUENCY DISTRIBUTION TABLES FOR 1994-1998 DATA WITH STANDARD M/S BINS

1 **1994-98 SURRY MET DATA** **10:38 WEDNESDAY,**
FEBRUARY 28, 2001 **1**

STABILITY CLASS A

TABLE OF WINSPEED BY SECTOR

WINSPEED **SECTOR**

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	0	0	0	0	1	0	0	0	1
B) 0.51-.75	0	0	0	0	0	0	0	1	3
C) 0.76-1.0	0	1	3	0	1	2	1	3	15
D) 1.1-1.50	9	7	8	18	23	34	50	26	328
E) 1.51-2.0	41	42	65	95	163	186	158	35	1400
F) 2.1-3.00	131	173	170	197	352	379	159	32	3335
G) 3.1-5.00	64	89	58	95	319	10	3	0	2046
H) 5.1-7.00	0	0	3	2	24	0	0	0	130
I) 7.1-10.0	0	0	0	0	0	0	0	0	3
TOTAL	245	312	307	407	883	611	371	97	7261

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED **SECTOR**

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	0	0	0	0	0	0	0	0	1
B) 0.51-.75	1	0	0	1	0	0	0	0	3
C) 0.76-1.0	1	1	0	0	0	2	0	0	15
D) 1.1-1.50	23	17	7	13	32	39	18	4	328
E) 1.51-2.0	35	49	38	43	135	174	95	46	1400
F) 2.1-3.00	41	128	260	157	229	407	371	149	3335
G) 3.1-5.00	6	69	370	156	98	208	300	201	2046
H) 5.1-7.00	1	1	20	8	6	1	49	15	130
I) 7.1-10.0	0	0	0	0	0	0	2	1	3
TOTAL	108	265	695	378	500	831	835	416	7261

STABILITY CLASS B

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
B) 0.51-.75	1	0	0	0	0	1	1	0	3
C) 0.76-1.0	0	0	0	1	2	3	6	0	16
D) 1.1-1.50	7	8	10	6	19	22	22	10	187
E) 1.51-2.0	13	21	19	20	48	47	19	7	367
F) 2.1-3.00	35	35	44	51	86	42	9	2	565
G) 3.1-5.00	13	10	11	12	53	4	0	3	328
H) 5.1-7.00	0	0	0	0	1	0	0	0	26
I) 7.1-10.0	0	0	0	0	2	0	0	0	3
TOTAL	69	74	84	90	211	119	57	22	1495

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
B) 0.51-.75	0	0	0	0	0	0	0	0	3
C) 0.76-1.0	2	0	1	0	0	1	0	0	16
D) 1.1-1.50	5	9	9	7	16	21	6	10	187
E) 1.51-2.0	8	20	11	18	34	34	29	19	367
F) 2.1-3.00	9	19	70	32	26	28	37	40	565
G) 3.1-5.00	2	17	62	45	14	9	31	42	328
H) 5.1-7.00	0	0	4	3	2	0	13	3	26
I) 7.1-10.0	0	0	0	0	0	0	1	0	3
TOTAL	26	65	157	105	92	93	117	114	1495

STABILITY CLASS C

TABLE OF WINSPEED BY SECTOR

WINSPEED FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
B) 0.51-.75	0	1	0	0	0	0	0	0	8
C) 0.76-1.0	3	1	1	1	2	4	5	6	49
D) 1.1-1.50	15	6	14	19	19	28	25	13	252
E) 1.51-2.0	29	33	29	27	45	39	19	7	385
F) 2.1-3.00	57	32	40	57	83	36	9	4	583
G) 3.1-5.00	20	15	23	13	62	0	0	2	374
H) 5.1-7.00	0	0	0	1	6	0	0	0	36
I) 7.1-10.0	0	0	0	0	1	0	0	0	1
TOTAL	124	88	107	118	218	107	58	32	1688

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
B) 0.51-.75	2	1	0	1	1	1	1	0	8
C) 0.76-1.0	3	0	3	4	8	5	1	2	49
D) 1.1-1.50	5	9	11	17	19	15	17	20	252
E) 1.51-2.0	6	14	17	11	23	38	28	20	385
F) 2.1-3.00	5	21	61	43	29	26	45	35	583
G) 3.1-5.00	5	13	62	43	16	12	36	52	374
H) 5.1-7.00	0	0	2	8	3	1	8	7	36
I) 7.1-10.0	0	0	0	0	0	0	0	0	1
TOTAL	26	58	156	127	99	98	136	136	1688

STABILITY CLASS D

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	6	4	3	2	2	5	8	4	61
B) 0.51-.75	3	6	7	6	5	12	16	12	143
C) 0.76-1.0	21	21	39	23	27	33	59	36	500
D) 1.1-1.50	105	94	89	96	89	115	128	56	1477
E) 1.51-2.0	167	164	149	114	178	161	93	33	1860
F) 2.1-3.00	440	277	257	248	435	122	64	26	3489
G) 3.1-5.00	184	111	104	116	427	7	9	19	2458
H) 5.1-7.00	1	3	5	13	53	0	0	2	239
I) 7.1-10.0	0	0	0	0	5	0	0	0	10
TOTAL	927	680	653	618	1221	455	377	188	10237

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	4	5	1	3	4	3	4	3	61
B) 0.51-.75	8	10	15	10	12	6	6	9	143
C) 0.76-1.0	27	28	33	36	37	40	22	18	500
D) 1.1-1.50	51	61	93	99	107	91	87	116	1477
E) 1.51-2.0	44	82	103	102	104	97	109	160	1860
F) 2.1-3.00	67	139	318	258	112	99	206	421	3489
G) 3.1-5.00	30	119	263	201	72	45	279	472	2458
H) 5.1-7.00	1	0	15	15	2	5	72	52	239
I) 7.1-10.0	0	0	0	1	0	0	4	0	10
TOTAL	232	444	841	725	450	386	789	1251	10237

1 1994-98 SURRY MET DATA 10:38 WEDNESDAY,
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STABILITY CLASS E

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	9	5	11	9	19	31	56	52	315
B) 0.51-.75	21	14	19	14	19	40	74	54	465
C) 0.76-1.0	61	44	57	48	51	154	187	134	1260
D) 1.1-1.50	199	113	163	104	118	357	280	220	2823
E) 1.51-2.0	212	75	151	125	158	224	171	111	2843
F) 2.1-3.00	191	103	81	102	242	105	121	61	3686
G) 3.1-5.00	58	24	15	18	147	4	15	13	1311
H) 5.1-7.00	0	1	11	9	19	1	0	0	88
I) 7.1-10.0	0	0	0	1	9	1	0	0	11
J) 10.1-13.	0	0	0	0	3	0	0	0	3
TOTAL	751	379	508	430	785	917	904	645	12805

(CONTINUED)

1 1994-98 SURRY MET DATA 10:38 WEDNESDAY,
FEBRUARY 28, 2001 6

STABILITY CLASS E

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	24	14	17	22	14	12	10	10	315
B) 0.51-.75	43	29	33	42	20	18	11	14	465
C) 0.76-1.0	100	80	94	93	55	21	28	53	1260
D) 1.1-1.50	176	191	222	263	123	66	77	151	2823
E) 1.51-2.0	157	253	366	330	151	51	83	225	2843
F) 2.1-3.00	149	388	783	489	190	87	219	375	3686
G) 3.1-5.00	50	157	213	180	62	40	155	160	1311
H) 5.1-7.00	1	0	7	10	2	3	14	10	88
I) 7.1-10.0	0	0	0	0	0	0	0	0	11
J) 10.1-13.	0	0	0	0	0	0	0	0	3
TOTAL	700	1112	1735	1429	617	298	597	998	12805

STABILITY CLASS F

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	11	19	28	27	40	68	128	100	671
B) 0.51-.75	17	20	18	16	24	55	149	78	661
C) 0.76-1.0	52	34	33	24	31	128	167	110	1089
D) 1.1-1.50	51	24	24	8	35	73	73	49	1069
E) 1.51-2.0	7	2	5	1	18	9	8	6	401
F) 2.1-3.00	1	1	0	1	3	0	2	0	123
G) 3.1-5.00	1	0	0	0	0	0	0	0	5
TOTAL	140	100	108	77	151	333	527	343	4019

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	58	32	31	24	36	27	22	20	671
B) 0.51-.75	59	50	32	37	32	26	20	28	661
C) 0.76-1.0	93	99	91	67	53	37	32	38	1089
D) 1.1-1.50	79	143	155	167	53	24	47	64	1069
E) 1.51-2.0	6	58	119	99	26	9	15	13	401
F) 2.1-3.00	0	8	56	31	7	0	8	5	123
G) 3.1-5.00	0	0	1	3	0	0	0	0	5
TOTAL	295	390	485	428	207	123	144	168	4019

STABILITY CLASS G

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	A) N	B) NNE	C) NE	D) ENE	E) E	F) ESE	G) SE	H) SSE	TOTAL
A) 0.0-0.50	160	74	84	74	154	166	254	249	3215
B) 0.51-.75	33	13	11	13	18	44	106	63	907
C) 0.76-1.0	31	12	3	11	15	25	36	23	753
D) 1.1-1.50	10	1	1	3	2	5	2	1	349
E) 1.51-2.0	1	0	0	1	0	1	0	0	34
F) 2.1-3.00	0	0	0	1	0	0	0	0	3
TOTAL	235	100	99	103	189	241	398	336	5261

(CONTINUED)

TABLE OF WINSPEED BY SECTOR

WINSPEED SECTOR

FREQUENCY	I) S	J) SSW	K) SW	L) WSW	M) W	N) WNW	O) NW	P) NNW	TOTAL
A) 0.0-0.50	261	194	198	179	253	332	377	206	3215
B) 0.51-.75	91	87	90	78	83	77	61	39	907
C) 0.76-1.0	47	145	155	75	59	40	38	38	753
D) 1.1-1.50	9	82	134	69	5	6	13	6	349
E) 1.51-2.0	1	1	14	10	1	3	0	1	34
F) 2.1-3.00	0	0	0	0	1	0	0	1	3
TOTAL	409	509	591	411	402	458	489	291	5261

TABLE 3. PAVAN Input File - Unit 1 Reactor Release
(1982-86 meteorological data)

```
0100 00001
Surry (VA Power)      1982-86 JFD        Ground-Level Release
  9.6 m               9.6 m - 44.9 m
Onsite
Unit 1 Ractor Release
    9    0
  960. 35.5  10.  9.6
   0    0    0    0    0    0    0
   1    1    0    2    0    0    0    0    1    0    0    0    1    2    3    1
   1    0    1    1    1    0    4    2    1    0    1    3    1    0    1    0
   1    1    0    7    5    6    4    3    5    7    2    0    1    1    2    4
  24   15   16   55   60   50   66   37   22   22   21   12   24   23   28   21
  69   41   57  171  171  173  173   72   44   55   49   55   97  128   94   71
 186  174  186  366  283  238  354   77   86  194  191  162  244  359  251  206
 293  232  181  289  201   12   45   15   47  290  437  220  137  323  274  287
  38    3    6    2    4    0    0    1    3   42   81   21   27   45   50   37
   0    0    0    2    0    0    0    0    0    1    3    2   11    5    6    1
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
   0    1    0    1    0    0    1    0    0    0    0    0    0    0    0    0
   0    2    0    0    1    2    5    0    1    0    3    2    3    1    2    0
  11    2    3   10   11   12   19    9   13    7    9   10    6   13    4    5
  16    8    9   25   18   23   23    7    9   15   15   13   21   18   20   14
  37   31   26   34   39   10   16    8    7   20   30   35   35   40   27   27
  58   37   28   27   18    2    5    6   12   34   49   34   17   27   20   21
   3    0    3    1    1    0    0    0    0    9    8    6    7    2    8    3
   0    0    0    2    0    0    0    0    0    1    1    0    0    2    0    0
   0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
   0    0    1    0    1    0    3    0    1    1    0    0    0    0    0    0
   2    1    0    3    3    1    4    3    1    2    4    1    0    1    1    0
   9    7    3    7   14   24   23   13    8   14    6    8    9   10    6    5
  16   14   12   17   24   13   13    9    8    9   20   12   15   15   18   13
  51   32   37   33   42   21   11    5    9   31   41   31   25   22   34   30
  45   40   35   29   16    1    1    6   13   40   59   39   26   23   19   37
   3    1    0    2    1    0    0    0    1    5   11    2    7    2    8   11
   1    0    0    0    0    0    0    0    0    1    0    0    1    0    1    1
   3    1    0    1    3    1    1    4    4    0    2    1    1    0    0    1
   2    1    0    7    3    5   11    7    6    5    5    5    3    5    5    2
  12   17   13   12   18   43   43   29   20   28   18   20   17   12   17   15
  54   44   61   52   88  112   79   70   48   62   52   52   50   46   35   53
  97   88  106   86  128  130   52   54   45   55   75   61   73   79   72  108
 277  261  293  228  252   93   86   45   72  167  160  146  118  114  139  253
 412  330  283  281  170   22   30   30   64  205  208  125  115  107  147  325
  75   36   26   54   12    6    0    4   19   47   40   22   27   32   91   70
   6    1    7   17    3    1    0    0    1    7    1    1    6    4    4    1
   3    2    1    5    8    7    8   14   11   10    7    2    3    0    2    5
   8    5   10    7    6   17   28   29   22   18    8    6   10    8    5    4
  21   14   16   18   37   62   97   94   70   47   29   31   23   25   18   20
  73   92   77   55  111  290  141  160  125  105  113  110   79   60   51   61
```

TABLE 3. (Cont'd) PAVAN Input File - Unit 1 Reactor Release
(1982-86 meteorological data)

148	130	151	103	238	294	85	136	181	124	142	137	102	89	67	112
417	265	243	122	186	117	95	160	354	596	411	294	186	131	178	310
224	123	115	50	60	16	46	43	201	592	327	187	147	111	187	276
19	1	9	9	9	5	1	3	12	33	30	18	20	19	26	14
10	3	5	3	7	0	0	0	0	1	4	1	6	0	1	0
4	5	7	3	8	12	28	19	21	12	11	7	3	3	4	5
7	5	6	8	18	23	44	49	25	21	12	5	8	6	6	13
22	12	16	27	19	69	117	138	91	63	40	31	24	23	23	34
111	64	36	36	41	107	110	198	199	117	103	82	85	64	69	100
79	31	15	14	19	11	23	77	197	191	167	106	82	54	44	55
27	12	3	4	1	1	1	25	135	278	163	91	63	27	21	13
8	1	0	0	0	0	0	2	7	21	15	2	4	1	0	4
1	0	0	0	0	0	0	1	0	0	0	0	1	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0
15	7	12	10	16	16	30	62	73	59	50	31	25	17	21	15
19	15	7	9	24	27	79	99	116	81	61	48	37	28	20	23
30	23	29	28	37	64	119	204	278	256	138	98	66	69	56	46
39	22	11	15	7	13	65	188	245	287	121	70	68	51	45	43
7	3	0	1	2	1	5	21	104	141	70	19	14	12	13	14
1	0	0	1	0	1	0	0	44	78	28	3	3	3	2	6
0	0	0	0	0	0	0	0	0	0	2	0	3	1	0	1
0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
-1.	0.50	0.75	1.0	1.5	2.0	3.0	5.0	7.0	10.0						
503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.
4827.	4827.	4827.	4827.	4827.	4827.	4827.	4827.	4827.	4827.	4827.	4827.	4827.	4827.	4827.	4827.

TABLE 3. (Cont'd) PAVAN Input File - Unit 2 Reactor Release
(1982-86 meteorological data)

```
0100 00001
Surry (VA Power)      1982-86 JFD        Ground-Level Release
  9.6 m               9.6 m - 44.9 m
Onsite
Unit 2 Ractor Release
    9    0
 960. 35.5  10.   9.6
    0    0    0    0    0    0    0
    1    1    0    2    0    0    0    0    1    0    0    0    1    2    3    1
    1    0    1    1    1    0    4    2    1    0    1    3    1    0    1    0
    1    1    0    7    5    6    4    3    5    7    2    0    1    1    2    4
   24   15   16   55   60   50   66   37   22   22   21   12   24   23   28   21
   69   41   57  171  171  173  173   72   44   55   49   55   97  128   94   71
  186  174  186  366  283  238  354   77   86  194  191  162  244  359  251  206
  293  232  181  289  201   12   45   15   47  290  437  220  137  323  274  287
   38    3    6    2    4    0    0    1    3   42   81   21   27   45   50   37
    0    0    0    2    0    0    0    0    0    1    3    2   11    5    6    1
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    1    0    1    0    0    1    0    0    0    0    0    0    0    0    0
    0    2    0    0    1    2    5    0    1    0    3    2    3    1    2    0
   11    2    3   10   11   12   19    9   13    7    9   10    6   13    4    5
   16    8    9   25   18   23   23    7    9   15   15   13   21   18   20   14
   37   31   26   34   39   10   16    8    7   20   30   35   35   40   27   27
   58   37   28   27   18    2    5    6   12   34   49   34   17   27   20   21
    3    0    3    1    1    0    0    0    0    9    8    6    7    2    8    3
    0    0    0    2    0    0    0    0    0    1    1    0    0    2    0    0
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    1    0    1    0    3    0    1    1    0    0    0    0    0    0
    2    1    0    3    3    1    4    3    1    2    4    1    0    1    1    0
    9    7    3    7   14   24   23   13    8   14    6    8    9   10    6    5
   16   14   12   17   24   13   13    9    8    9   20   12   15   15   18   13
   51   32   37   33   42   21   11    5    9   31   41   31   25   22   34   30
   45   40   35   29   16    1    1    6   13   40   59   39   26   23   19   37
    3    1    0    2    1    0    0    0    1    5   11    2    7    2    8   11
    1    0    0    0    0    0    0    0    0    1    0    0    1    0    1    1
    3    1    0    1    3    1    1    4    4    0    2    1    1    0    0    1
    2    1    0    7    3    5   11    7    6    5    5    5    3    5    5    2
   12   17   13   12   18   43   43   29   20   28   18   20   17   12   17   15
   54   44   61   52   88  112   79   70   48   62   52   52   50   46   35   53
   97   88  106   86  128  130   52   54   45   55   75   61   73   79   72  108
  277  261  293  228  252   93   86   45   72  167  160  146  118  114  139  253
  412  330  283  281  170   22   30   30   64  205  208  125  115  107  147  325
   75   36   26   54   12    6    0    4   19   47   40   22   27   32   91   70
    6    1    7   17    3    1    0    0    1    7    1    1    6    4    4    1
    3    2    1    5    8    7    8   14   11   10    7    2    3    0    2    5
    8    5   10    7    6   17   28   29   22   18    8    6   10    8    5    4
   21   14   16   18   37   62   97   94   70   47   29   31   23   25   18   20
   73   92   77   55  111  290  141  160  125  105  113  110   79   60   51   61
  148  130  151  103  238  294   85  136  181  124  142  137  102   89   67  112
```

TABLE 3. (Cont'd) PAVAN Input File - Unit 2 Reactor Release
(1982-86 meteorological data)

417	265	243	122	186	117	95	160	354	596	411	294	186	131	178	310
224	123	115	50	60	16	46	43	201	592	327	187	147	111	187	276
19	1	9	9	9	5	1	3	12	33	30	18	20	19	26	14
10	3	5	3	7	0	0	0	0	1	4	1	6	0	1	0
4	5	7	3	8	12	28	19	21	12	11	7	3	3	4	5
7	5	6	8	18	23	44	49	25	21	12	5	8	6	6	13
22	12	16	27	19	69	117	138	91	63	40	31	24	23	23	34
111	64	36	36	41	107	110	198	199	117	103	82	85	64	69	100
79	31	15	14	19	11	23	77	197	191	167	106	82	54	44	55
27	12	3	4	1	1	1	25	135	278	163	91	63	27	21	13
8	1	0	0	0	0	0	2	7	21	15	2	4	1	0	4
1	0	0	0	0	0	0	1	0	0	0	0	1	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0
15	7	12	10	16	16	30	62	73	59	50	31	25	17	21	15
19	15	7	9	24	27	79	99	116	81	61	48	37	28	20	23
30	23	29	28	37	64	119	204	278	256	138	98	66	69	56	46
39	22	11	15	7	13	65	188	245	287	121	70	68	51	45	43
7	3	0	1	2	1	5	21	104	141	70	19	14	12	13	14
1	0	0	1	0	1	0	0	44	78	28	3	3	3	2	6
0	0	0	0	0	0	0	0	0	0	2	0	3	1	0	1
0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
-1.	0.50	0.75	1.0	1.5	2.0	3.0	5.0	7.0	10.0						
503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.
4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.

TABLE 3. (Cont'd) PAVAN Input File - Vent Stack #2 Release
(1982-86 meteorological data)

```
0100 00001
Surry (VA Power)     1982-86 JFD        Ground-Level Release
  9.6 m              9.6 m - 44.9 m
Onsite
Vent Release (no plume rise)
    9    0
 960. 35.5  10.  9.6
    0    0    0    0    0    0    0
    1    1    0    2    0    0    0    0    1    0    0    0    1    2    3    1
    1    0    1    1    1    0    4    2    1    0    1    3    1    0    1    0
    1    1    0    7    5    6    4    3    5    7    2    0    1    1    2    4
   24   15   16   55   60   50   66   37   22   22   21   12   24   23   28   21
   69   41   57  171  171  173  173   72   44   55   49   55   97  128   94   71
  186  174  186  366  283  238  354   77   86  194  191  162  244  359  251  206
  293  232  181  289  201   12   45   15   47  290  437  220  137  323  274  287
   38    3    6    2    4    0    0    1    3   42   81   21   27   45   50   37
    0    0    0    2    0    0    0    0    0    1    3    2   11    5    6    1
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    1    0    1    0    0    1    0    0    0    0    0    0    0    0    0
    0    2    0    0    1    2    5    0    1    0    3    2    3    1    2    0
   11    2    3   10   11   12   19    9   13    7    9   10    6   13    4    5
   16    8    9   25   18   23   23    7    9   15   15   13   21   18   20   14
   37   31   26   34   39   10   16    8    7   20   30   35   35   40   27   27
   58   37   28   27   18    2    5    6   12   34   49   34   17   27   20   21
    3    0    3    1    1    0    0    0    0    9    8    6    7    2    8    3
    0    0    0    2    0    0    0    0    0    1    1    0    0    2    0    0
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    1    0    1    0    3    0    1    1    0    0    0    0    0    0
    2    1    0    3    3    1    4    3    1    2    4    1    0    1    1    0
    9    7    3    7   14   24   23   13    8   14    6    8    9   10    6    5
   16   14   12   17   24   13   13    9    8    9   20   12   15   15   18   13
   51   32   37   33   42   21   11    5    9   31   41   31   25   22   34   30
   45   40   35   29   16    1    1    6   13   40   59   39   26   23   19   37
    3    1    0    2    1    0    0    0    1    5   11    2    7    2    8   11
    1    0    0    0    0    0    0    0    0    1    0    0    1    0    1    1
    3    1    0    1    3    1    1    4    4    0    2    1    1    0    0    1
    2    1    0    7    3    5   11    7    6    5    5    5    3    5    5    2
   12   17   13   12   18   43   43   29   20   28   18   20   17   12   17   15
   54   44   61   52   88  112   79   70   48   62   52   52   50   46   35   53
   97   88  106   86  128  130   52   54   45   55   75   61   73   79   72  108
  277  261  293  228  252   93   86   45   72  167  160  146  118  114  139  253
  412  330  283  281  170   22   30   30   64  205  208  125  115  107  147  325
   75   36   26   54   12    6    0    4   19   47   40   22   27   32   91   70
    6    1    7   17    3    1    0    0    1    7    1    1    6    4    4    1
    3    2    1    5    8    7    8   14   11   10    7    2    3    0    2    5
    8    5   10    7    6   17   28   29   22   18    8    6   10    8    5    4
   21   14   16   18   37   62   97   94   70   47   29   31   23   25   18   20
   73   92   77   55  111  290  141  160  125  105  113  110   79   60   51   61
```

TABLE 3. (Cont'd) PAVAN Input File - Vent Stack #2 Release
(1982-86 meteorological data)

148	130	151	103	238	294	85	136	181	124	142	137	102	89	67	112
417	265	243	122	186	117	95	160	354	596	411	294	186	131	178	310
224	123	115	50	60	16	46	43	201	592	327	187	147	111	187	276
19	1	9	9	9	5	1	3	12	33	30	18	20	19	26	14
10	3	5	3	7	0	0	0	0	1	4	1	6	0	1	0
4	5	7	3	8	12	28	19	21	12	11	7	3	3	4	5
7	5	6	8	18	23	44	49	25	21	12	5	8	6	6	13
22	12	16	27	19	69	117	138	91	63	40	31	24	23	23	34
111	64	36	36	41	107	110	198	199	117	103	82	85	64	69	100
79	31	15	14	19	11	23	77	197	191	167	106	82	54	44	55
27	12	3	4	1	1	1	25	135	278	163	91	63	27	21	13
8	1	0	0	0	0	0	2	7	21	15	2	4	1	0	4
1	0	0	0	0	0	0	1	0	0	0	0	1	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0
15	7	12	10	16	16	30	62	73	59	50	31	25	17	21	15
19	15	7	9	24	27	79	99	116	81	61	48	37	28	20	23
30	23	29	28	37	64	119	204	278	256	138	98	66	69	56	46
39	22	11	15	7	13	65	188	245	287	121	70	68	51	45	43
7	3	0	1	2	1	5	21	104	141	70	19	14	12	13	14
1	0	0	1	0	1	0	0	44	78	28	3	3	3	2	6
0	0	0	0	0	0	0	0	0	0	2	0	3	1	0	1
0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
-1.	0.50	0.75	1.0	1.5	2.0	3.0	5.0	7.0	10.0						
503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.
4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.

TABLE 3. (Cont'd) PAVAN Input File - Aux. Bldg. West Louver Release
(1982-86 meteorological data)

```
0100 00001
Surry (VA Power)      1982-86 JFD         Ground-Level Release
  9.6 m               9.6 m - 44.9 m
Onsite
Aux Bldg West Louver Release
    9    0
 960. 35.5  10.  9.6
    0    0    0    0    0    0    0
    1    1    0    2    0    0    0    0    1    0    0    0    1    2    3    1
    1    0    1    1    1    0    4    2    1    0    1    3    1    0    1    0
    1    1    0    7    5    6    4    3    5    7    2    0    1    1    2    4
   24   15   16   55   60   50   66   37   22   22   21   12   24   23   28   21
   69   41   57  171  171  173  173   72   44   55   49   55   97  128   94   71
  186  174  186  366  283  238  354   77   86  194  191  162  244  359  251  206
  293  232  181  289  201   12   45   15   47  290  437  220  137  323  274  287
   38    3    6    2    4    0    0    1    3   42   81   21   27   45   50   37
    0    0    0    2    0    0    0    0    0    1    3    2   11    5    6    1
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    1    0    1    0    0    1    0    0    0    0    0    0    0    0    0
    0    2    0    0    1    2    5    0    1    0    3    2    3    1    2    0
   11    2    3   10   11   12   19    9   13    7    9   10    6   13    4    5
   16    8    9   25   18   23   23    7    9   15   15   13   21   18   20   14
   37   31   26   34   39   10   16    8    7   20   30   35   35   40   27   27
   58   37   28   27   18    2    5    6   12   34   49   34   17   27   20   21
    3    0    3    1    1    0    0    0    0    9    8    6    7    2    8    3
    0    0    0    2    0    0    0    0    0    1    1    0    0    2    0    0
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    1    0    1    0    3    0    1    1    0    0    0    0    0    0
    2    1    0    3    3    1    4    3    1    2    4    1    0    1    1    0
    9    7    3    7   14   24   23   13    8   14    6    8    9   10    6    5
   16   14   12   17   24   13   13    9    8    9   20   12   15   15   18   13
   51   32   37   33   42   21   11    5    9   31   41   31   25   22   34   30
   45   40   35   29   16    1    1    6   13   40   59   39   26   23   19   37
    3    1    0    2    1    0    0    0    1    5   11    2    7    2    8   11
    1    0    0    0    0    0    0    0    0    1    0    0    1    0    1    1
    3    1    0    1    3    1    1    4    4    0    2    1    1    0    0    1
    2    1    0    7    3    5   11    7    6    5    5    5    3    5    5    2
   12   17   13   12   18   43   43   29   20   28   18   20   17   12   17   15
   54   44   61   52   88  112   79   70   48   62   52   52   50   46   35   53
   97   88  106   86  128  130   52   54   45   55   75   61   73   79   72  108
  277  261  293  228  252   93   86   45   72  167  160  146  118  114  139  253
  412  330  283  281  170   22   30   30   64  205  208  125  115  107  147  325
   75   36   26   54   12    6    0    4   19   47   40   22   27   32   91   70
    6    1    7   17    3    1    0    0    1    7    1    1    6    4    4    1
    3    2    1    5    8    7    8   14   11   10    7    2    3    0    2    5
    8    5   10    7    6   17   28   29   22   18    8    6   10    8    5    4
   21   14   16   18   37   62   97   94   70   47   29   31   23   25   18   20
   73   92   77   55  111  290  141  160  125  105  113  110   79   60   51   61
```

TABLE 3. (Cont'd) PAVAN Input File - Aux. Bldg. West Louver Release
(1982-86 meteorological data)

```
 417  265  243  122  186  117   95  160  354  596  411  294  186  131  178  310
 224  123  115   50   60   16   46   43  201  592  327  187  147  111  187  276
  19    1    9    9    9    5    1    3   12   33   30   18   20   19   26   14
  10    3    5    3    7    0    0    0    0    1    4    1    6    0    1    0
   4    5    7    3    8   12   28   19   21   12   11    7    3    3    4    5
   7    5    6    8   18   23   44   49   25   21   12    5    8    6    6   13
  22   12   16   27   19   69  117  138   91   63   40   31   24   23   23   34
 111   64   36   36   41  107  110  198  199  117  103   82   85   64   69  100
  79   31   15   14   19   11   23   77  197  191  167  106   82   54   44   55
  27   12    3    4    1    1    1   25  135  278  163   91   63   27   21   13
   8    1    0    0    0    0    0    2    7   21   15    2    4    1    0    4
   1    0    0    0    0    0    0    1    0    0    0    0    1    0    0    0
   0    0    0    0    0    0    0    0    0    0    0    0    1    0    0    0
  15    7   12   10   16   16   30   62   73   59   50   31   25   17   21   15
  19   15    7    9   24   27   79   99  116   81   61   48   37   28   20   23
  30   23   29   28   37   64  119  204  278  256  138   98   66   69   56   46
  39   22   11   15    7   13   65  188  245  287  121   70   68   51   45   43
   7    3    0    1    2    1    5   21  104  141   70   19   14   12   13   14
   1    0    0    1    0    1    0    0   44   78   28    3    3    3    2    6
   0    0    0    0    0    0    0    0    0    0    2    0    3    1    0    1
   0    0    0    0    0    0    0    0    0    0    0    1    0    0    0    1
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
 -1. 0.50 0.75  1.0  1.5  2.0  3.0  5.0  7.0 10.0
 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503.
4793.4793.4793.4793.4793.4793.4793.4793.4793.4793.4793.4793.4793.4793.4793.4793.
```

TABLE 3. (Cont'd) PAVAN Input File - Aux. Bldg. East Louver Release
(1982-86 meteorological data)

```
0100 00001
Surry (VA Power)     1982-86 JFD        Ground-Level Release
  9.6 m              9.6 m - 44.9 m
Onsite
Aux Bldg East Louver Release
    9    0
 960. 35.5  10.   9.6
   0    0    0    0    0    0    0
   1    1    0    2    0    0    0    0    1    0    0    0    1    2    3    1
   1    0    1    1    1    0    4    2    1    0    1    3    1    0    1    0
   1    1    0    7    5    6    4    3    5    7    2    0    1    1    2    4
  24   15   16   55   60   50   66   37   22   22   21   12   24   23   28   21
  69   41   57  171  171  173  173   72   44   55   49   55   97  128   94   71
 186  174  186  366  283  238  354   77   86  194  191  162  244  359  251  206
 293  232  181  289  201   12   45   15   47  290  437  220  137  323  274  287
  38    3    6    2    4    0    0    1    3   42   81   21   27   45   50   37
   0    0    0    2    0    0    0    0    0    1    3    2   11    5    6    1
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
   0    1    0    1    0    0    1    0    0    0    0    0    0    0    0    0
   0    2    0    0    1    2    5    0    1    0    3    2    3    1    2    0
  11    2    3   10   11   12   19    9   13    7    9   10    6   13    4    5
  16    8    9   25   18   23   23    7    9   15   15   13   21   18   20   14
  37   31   26   34   39   10   16    8    7   20   30   35   35   40   27   27
  58   37   28   27   18    2    5    6   12   34   49   34   17   27   20   21
   3    0    3    1    1    0    0    0    0    9    8    6    7    2    8    3
   0    0    0    2    0    0    0    0    0    1    1    0    0    2    0    0
   0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
   0    0    1    0    1    0    3    0    1    1    0    0    0    0    0    0
   2    1    0    3    3    1    4    3    1    2    4    1    0    1    1    0
   9    7    3    7   14   24   23   13    8   14    6    8    9   10    6    5
  16   14   12   17   24   13   13    9    8    9   20   12   15   15   18   13
  51   32   37   33   42   21   11    5    9   31   41   31   25   22   34   30
  45   40   35   29   16    1    1    6   13   40   59   39   26   23   19   37
   3    1    0    2    1    0    0    0    1    5   11    2    7    2    8   11
   1    0    0    0    0    0    0    0    0    1    0    0    1    0    1    1
   3    1    0    1    3    1    1    4    4    0    2    1    1    0    0    1
   2    1    0    7    3    5   11    7    6    5    5    5    3    5    5    2
  12   17   13   12   18   43   43   29   20   28   18   20   17   12   17   15
  54   44   61   52   88  112   79   70   48   62   52   52   50   46   35   53
  97   88  106   86  128  130   52   54   45   55   75   61   73   79   72  108
 277  261  293  228  252   93   86   45   72  167  160  146  118  114  139  253
 412  330  283  281  170   22   30   30   64  205  208  125  115  107  147  325
  75   36   26   54   12    6    0    4   19   47   40   22   27   32   91   70
   6    1    7   17    3    1    0    0    1    7    1    1    6    4    4    1
   3    2    1    5    8    7    8   14   11   10    7    2    3    0    2    5
   8    5   10    7    6   17   28   29   22   18    8    6   10    8    5    4
  21   14   16   18   37   62   97   94   70   47   29   31   23   25   18   20
  73   92   77   55  111  290  141  160  125  105  113  110   79   60   51   61
```

TABLE 3. (Cont'd) PAVAN Input File - Aux. Bldg. East Louver Release
(1982-86 meteorological data)

417	265	243	122	186	117	95	160	354	596	411	294	186	131	178	310
224	123	115	50	60	16	46	43	201	592	327	187	147	111	187	276
19	1	9	9	9	5	1	3	12	33	30	18	20	19	26	14
10	3	5	3	7	0	0	0	0	1	4	1	6	0	1	0
4	5	7	3	8	12	28	19	21	12	11	7	3	3	4	5
7	5	6	8	18	23	44	49	25	21	12	5	8	6	6	13
22	12	16	27	19	69	117	138	91	63	40	31	24	23	23	34
111	64	36	36	41	107	110	198	199	117	103	82	85	64	69	100
79	31	15	14	19	11	23	77	197	191	167	106	82	54	44	55
27	12	3	4	1	1	1	25	135	278	163	91	63	27	21	13
8	1	0	0	0	0	0	2	7	21	15	2	4	1	0	4
1	0	0	0	0	0	0	1	0	0	0	0	1	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	0
15	7	12	10	16	16	30	62	73	59	50	31	25	17	21	15
19	15	7	9	24	27	79	99	116	81	61	48	37	28	20	23
30	23	29	28	37	64	119	204	278	256	138	98	66	69	56	46
39	22	11	15	7	13	65	188	245	287	121	70	68	51	45	43
7	3	0	1	2	1	5	21	104	141	70	19	14	12	13	14
1	0	0	1	0	1	0	0	44	78	28	3	3	3	2	6
0	0	0	0	0	0	0	0	0	0	2	0	3	1	0	1
0	0	0	0	0	0	0	0	0	0	0	1	0	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
-1.	0.50	0.75	1.0	1.5	2.0	3.0	5.0	7.0	10.0						
503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.
4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.

TABLE 3. (Cont'd) PAVAN Input File - Unit 1 Reactor Release
(1994-98 meteorological data)

```
0100 00001
Surry (VA Power)     1994-98 JFD        Ground-Level Release
  9.6 m              9.6 m - 44.9 m
Onsite
Unit 1 Ractor Release
  10    0
 960. 35.5  10.  9.6
   0    0    0    0    0    0    0
   0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
   0    0    0    0    0    0    0    1    1    0    0    1    0    0    0    0
   0    1    3    0    1    2    1    3    1    1    0    0    0    2    0    0
   9    7    8   18   23   34   50   26   23   17    7   13   32   39   18    4
  41   42   65   95  163  186  158   35   35   49   38   43  135  174   95   46
 131  173  170  197  352  379  159   32   41  128  260  157  229  407  371  149
  64   89   58   95  319   10    3    0    6   69  370  156   98  208  300  201
   0    0    3    2   24    0    0    0    1    1   20    8    6    1   49   15
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    2    1
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
   1    0    0    0    0    1    1    0    0    0    0    0    0    0    0    0
   0    0    0    1    2    3    6    0    2    0    1    0    0    1    0    0
   7    8   10    6   19   22   22   10    5    9    9    7   16   21    6   10
  13   21   19   20   48   47   19    7    8   20   11   18   34   34   29   19
  35   35   44   51   86   42    9    2    9   19   70   32   26   28   37   40
  13   10   11   12   53    4    0    3    2   17   62   45   14    9   31   42
   0    0    0    0    1    0    0    0    0    0    4    3    2    0   13    3
   0    0    0    0    2    0    0    0    0    0    0    0    0    0    1    0
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
   0    1    0    0    0    0    0    0    2    1    0    1    1    1    1    0
   3    1    1    1    2    4    5    6    3    0    3    4    8    5    1    2
  15    6   14   19   19   28   25   13    5    9   11   17   19   15   17   20
  29   33   29   27   45   39   19    7    6   14   17   11   23   38   28   20
  57   32   40   57   83   36    9    4    5   21   61   43   29   26   45   35
  20   15   23   13   62    0    0    2    5   13   62   43   16   12   36   52
   0    0    0    1    6    0    0    0    0    0    2    8    3    1    8    7
   0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
   6    4    3    2    2    5    8    4    4    5    1    3    4    3    4    3
   3    6    7    6    5   12   16   12    8   10   15   10   12    6    6    9
  21   21   39   23   27   33   59   36   27   28   33   36   37   40   22   18
 105   94   89   96   89  115  128   56   51   61   93   99  107   91   87  116
 167  164  149  114  178  161   93   33   44   82  103  102  104   97  109  160
 440  277  257  248  435  122   64   26   67  139  318  258  112   99  206  421
 184  111  104  116  427    7    9   19   30  119  263  201   72   45  279  472
   1    3    5   13   53    0    0    2    1    0   15   15    2    5   72   52
   0    0    0    0    5    0    0    0    0    0    0    1    0    0    4    0
   0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
```

TABLE 3. (Cont'd) PAVAN Input File - Unit 1 Reactor Release
(1994-98 meteorological data)

```
   9    5   11    9   19   31   56   52  24   14   17   22   14   12   10   10
  21   14   19   14   19   40   74   54  43   29   33   42   20   18   11   14
  61   44   57   48   51  154  187  134 100   80   94   93   55   21   28   53
 199  113  163  104  118  357  280  220 176  191  222  263  123   66   77  151
 212   75  151  125  158  224  171  111 157  253  366  330  151   51   83  225
 191  103   81  102  242  105  121   61 149  388  783  489  190   87  219  375
  58   24   15   18  147    4   15   13  50  157  213  180   62   40  155  160
   0    1   11    9   19    1    0    0   1    0    7   10    2    3   14   10
   0    0    0    1    9    1    0    0   0    0    0    0    0    0    0    0
   0    0    0    0    3    0    0    0   0    0    0    0    0    0    0    0
  11   19   28   27   40   68  128  100  58   32   31   24   36   27   22   20
  17   20   18   16   24   55  149   78  59   50   32   37   32   26   20   28
  52   34   33   24   31  128  167  110  93   99   91   67   53   37   32   38
  51   24   24    8   35   73   73   49  79  143  155  167   53   24   47   64
   7    2    5    1   18    9    8    6   6   58  119   99   26    9   15   13
   1    1    0    1    3    0    2    0   0    8   56   31    7    0    8    5
   1    0    0    0    0    0    0    0   0    0    1    3    0    0    0    0
   0    0    0    0    0    0    0    0   0    0    0    0    0    0    0    0
   0    0    0    0    0    0    0    0   0    0    0    0    0    0    0    0
   0    0    0    0    0    0    0    0   0    0    0    0    0    0    0    0
 160   74   84   74  154  166  254  249 261  194  198  179  253  332  377  206
  33   13   11   13   18   44  106   63  91   87   90   78   83   77   61   39
  31   12    3   11   15   25   36   23  47  145  155   75   59   40   38   38
  10    1    1    3    2    5    2    1   9   82  134   69    5    6   13    6
   1    0    0    1    0    1    0    0   1    1   14   10    1    3    0    1
   0    0    0    1    0    0    0    0   0    0    0    0    1    0    0    1
   0    0    0    0    0    0    0    0   0    0    0    0    0    0    0    0
   0    0    0    0    0    0    0    0   0    0    0    0    0    0    0    0
   0    0    0    0    0    0    0    0   0    0    0    0    0    0    0    0
   0    0    0    0    0    0    0    0   0    0    0    0    0    0    0    0
 -1. 0.50 0.75  1.0  1.5  2.0  3.0  5.0  7.0 10.0 13.0
503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503. 503.
4827.4827.4827.4827.4827.4827.4827.4827.4827.4827.4827.4827.4827.4827.4827.4827.
```

TABLE 3. (Cont'd) PAVAN Input File - Unit 2 Reactor Release
(1994-98 meteorological data)

```
0100 00001
Surry (VA Power)      1994-98 JFD          Ground-Level Release
  9.6 m               9.6 m - 44.9 m
Onsite
Unit 2 Ractor Release
   10    0
  960. 35.5  10.  9.6
    0    0    0    0    0    0    0
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    1    1    0    0    1    0    0    0    0
    0    1    3    0    1    2    1    3    1    1    0    0    0    2    0    0
    9    7    8   18   23   34   50   26   23   17    7   13   32   39   18    4
   41   42   65   95  163  186  158   35   35   49   38   43  135  174   95   46
  131  173  170  197  352  379  159   32   41  128  260  157  229  407  371  149
   64   89   58   95  319   10    3    0    6   69  370  156   98  208  300  201
    0    0    3    2   24    0    0    0    1    1   20    8    6    1   49   15
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    2    1
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    1    0    0    0    0    1    1    0    0    0    0    0    0    0    0    0
    0    0    0    1    2    3    6    0    2    0    1    0    0    1    0    0
    7    8   10    6   19   22   22   10    5    9    9    7   16   21    6   10
   13   21   19   20   48   47   19    7    8   20   11   18   34   34   29   19
   35   35   44   51   86   42    9    2    9   19   70   32   26   28   37   40
   13   10   11   12   53    4    0    3    2   17   62   45   14    9   31   42
    0    0    0    0    1    0    0    0    0    0    4    3    2    0   13    3
    0    0    0    0    2    0    0    0    0    0    0    0    0    0    1    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    1    0    0    0    0    0    0    2    1    0    1    1    1    1    0
    3    1    1    1    2    4    5    6    3    0    3    4    8    5    1    2
   15    6   14   19   19   28   25   13    5    9   11   17   19   15   17   20
   29   33   29   27   45   39   19    7    6   14   17   11   23   38   28   20
   57   32   40   57   83   36    9    4    5   21   61   43   29   26   45   35
   20   15   23   13   62    0    0    2    5   13   62   43   16   12   36   52
    0    0    0    1    6    0    0    0    0    0    2    8    3    1    8    7
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    6    4    3    2    2    5    8    4    4    5    1    3    4    3    4    3
    3    6    7    6    5   12   16   12    8   10   15   10   12    6    6    9
   21   21   39   23   27   33   59   36   27   28   33   36   37   40   22   18
  105   94   89   96   89  115  128   56   51   61   93   99  107   91   87  116
  167  164  149  114  178  161   93   33   44   82  103  102  104   97  109  160
  440  277  257  248  435  122   64   26   67  139  318  258  112   99  206  421
  184  111  104  116  427    7    9   19   30  119  263  201   72   45  279  472
    1    3    5   13   53    0    0    2    1    0   15   15    2    5   72   52
    0    0    0    0    5    0    0    0    0    0    0    1    0    0    4    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
```

TABLE 3. (Cont'd) PAVAN Input File - Unit 2 Reactor Release
(1994-98 meteorological data)

9	5	11	9	19	31	56	52	24	14	17	22	14	12	10	10
21	14	19	14	19	40	74	54	43	29	33	42	20	18	11	14
61	44	57	48	51	154	187	134	100	80	94	93	55	21	28	53
199	113	163	104	118	357	280	220	176	191	222	263	123	66	77	151
212	75	151	125	158	224	171	111	157	253	366	330	151	51	83	225
191	103	81	102	242	105	121	61	149	388	783	489	190	87	219	375
58	24	15	18	147	4	15	13	50	157	213	180	62	40	155	160
0	1	11	9	19	1	0	0	1	0	7	10	2	3	14	10
0	0	0	1	9	1	0	0	0	0	0	0	0	0	0	0
0	0	0	0	3	0	0	0	0	0	0	0	0	0	0	0
11	19	28	27	40	68	128	100	58	32	31	24	36	27	22	20
17	20	18	16	24	55	149	78	59	50	32	37	32	26	20	28
52	34	33	24	31	128	167	110	93	99	91	67	53	37	32	38
51	24	24	8	35	73	73	49	79	143	155	167	53	24	47	64
7	2	5	1	18	9	8	6	6	58	119	99	26	9	15	13
1	1	0	1	3	0	2	0	0	8	56	31	7	0	8	5
1	0	0	0	0	0	0	0	0	0	1	3	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
160	74	84	74	154	166	254	249	261	194	198	179	253	332	377	206
33	13	11	13	18	44	106	63	91	87	90	78	83	77	61	39
31	12	3	11	15	25	36	23	47	145	155	75	59	40	38	38
10	1	1	3	2	5	2	1	9	82	134	69	5	6	13	6
1	0	0	1	0	1	0	0	1	1	14	10	1	3	0	1
0	0	0	1	0	0	0	0	0	0	0	0	1	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
-1.	0.50	0.75	1.0	1.5	2.0	3.0	5.0	7.0	10.0	13.0					
503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.
4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.	4744.

TABLE 3. (Cont'd) PAVAN Input File - Vent Stack #2 Release
(1994-98 meteorological data)

```
0100 00001
Surry (VA Power)      1994-98 JFD         Ground-Level Release
  9.6 m               9.6 m - 44.9 m
Onsite
Vent Release (no plume rise)
   10    0
  960. 35.5  10.  9.6
    0    0    0    0    0    0    0
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    1    1    0    0    1    0    0    0    0
    0    1    3    0    1    2    1    3    1    1    0    0    0    2    0    0
    9    7    8   18   23   34   50   26   23   17    7   13   32   39   18    4
   41   42   65   95  163  186  158   35   35   49   38   43  135  174   95   46
  131  173  170  197  352  379  159   32   41  128  260  157  229  407  371  149
   64   89   58   95  319   10    3    0    6   69  370  156   98  208  300  201
    0    0    3    2   24    0    0    0    1    1   20    8    6    1   49   15
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    2    1
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    1    0    0    0    0    1    1    0    0    0    0    0    0    0    0    0
    0    0    0    1    2    3    6    0    2    0    1    0    0    1    0    0
    7    8   10    6   19   22   22   10    5    9    9    7   16   21    6   10
   13   21   19   20   48   47   19    7    8   20   11   18   34   34   29   19
   35   35   44   51   86   42    9    2    9   19   70   32   26   28   37   40
   13   10   11   12   53    4    0    3    2   17   62   45   14    9   31   42
    0    0    0    0    1    0    0    0    0    0    4    3    2    0   13    3
    0    0    0    0    2    0    0    0    0    0    0    0    0    0    1    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    1    0    0    0    0    0    0    2    1    0    1    1    1    1    0
    3    1    1    1    2    4    5    6    3    0    3    4    8    5    1    2
   15    6   14   19   19   28   25   13    5    9   11   17   19   15   17   20
   29   33   29   27   45   39   19    7    6   14   17   11   23   38   28   20
   57   32   40   57   83   36    9    4    5   21   61   43   29   26   45   35
   20   15   23   13   62    0    0    2    5   13   62   43   16   12   36   52
    0    0    0    1    6    0    0    0    0    0    2    8    3    1    8    7
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    6    4    3    2    2    5    8    4    4    5    1    3    4    3    4    3
    3    6    7    6    5   12   16   12    8   10   15   10   12    6    6    9
   21   21   39   23   27   33   59   36   27   28   33   36   37   40   22   18
  105   94   89   96   89  115  128   56   51   61   93   99  107   91   87  116
  167  164  149  114  178  161   93   33   44   82  103  102  104   97  109  160
  440  277  257  248  435  122   64   26   67  139  318  258  112   99  206  421
  184  111  104  116  427    7    9   19   30  119  263  201   72   45  279  472
    1    3    5   13   53    0    0    2    1    0   15   15    2    5   72   52
    0    0    0    0    5    0    0    0    0    0    0    1    0    0    4    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
```

TABLE 3. (Cont'd) PAVAN Input File - Vent Stack #2 Release
(1994-98 meteorological data)

9	5	11	9	19	31	56	52	24	14	17	22	14	12	10	10
21	14	19	14	19	40	74	54	43	29	33	42	20	18	11	14
61	44	57	48	51	154	187	134	100	80	94	93	55	21	28	53
199	113	163	104	118	357	280	220	176	191	222	263	123	66	77	151
212	75	151	125	158	224	171	111	157	253	366	330	151	51	83	225
191	103	81	102	242	105	121	61	149	388	783	489	190	87	219	375
58	24	15	18	147	4	15	13	50	157	213	180	62	40	155	160
0	1	11	9	19	1	0	0	1	0	7	10	2	3	14	10
0	0	0	1	9	1	0	0	0	0	0	0	0	0	0	0
0	0	0	0	3	0	0	0	0	0	0	0	0	0	0	0
11	19	28	27	40	68	128	100	58	32	31	24	36	27	22	20
17	20	18	16	24	55	149	78	59	50	32	37	32	26	20	28
52	34	33	24	31	128	167	110	93	99	91	67	53	37	32	38
51	24	24	8	35	73	73	49	79	143	155	167	53	24	47	64
7	2	5	1	18	9	8	6	6	58	119	99	26	9	15	13
1	1	0	1	3	0	2	0	0	8	56	31	7	0	8	5
1	0	0	0	0	0	0	0	0	0	1	3	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
160	74	84	74	154	166	254	249	261	194	198	179	253	332	377	206
33	13	11	13	18	44	106	63	91	87	90	78	83	77	61	39
31	12	3	11	15	25	36	23	47	145	155	75	59	40	38	38
10	1	1	3	2	5	2	1	9	82	134	69	5	6	13	6
1	0	0	1	0	1	0	0	1	1	14	10	1	3	0	1
0	0	0	1	0	0	0	0	0	0	0	0	1	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
-1.	0.50	0.75	1.0	1.5	2.0	3.0	5.0	7.0	10.0	13.0					
503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.
4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.	4721.

TABLE 3. (Cont'd) PAVAN Input File - Aux. Bldg. West Release
(1994-98 meteorological data)

```
0100 00001
Surry (VA Power)     1994-98 JFD          Ground-Level Release
  9.6 m              9.6 m - 44.9 m
Onsite
Aux Bldg West Louver Release
   10    0
 960. 35.5  10.   9.6
    0    0    0    0    0    0    0
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    1    1    0    0    1    0    0    0    0
    0    1    3    0    1    2    1    3    1    1    0    0    0    2    0    0
    9    7    8   18   23   34   50   26   23   17    7   13   32   39   18    4
   41   42   65   95  163  186  158   35   35   49   38   43  135  174   95   46
  131  173  170  197  352  379  159   32   41  128  260  157  229  407  371  149
   64   89   58   95  319   10    3    0    6   69  370  156   98  208  300  201
    0    0    3    2   24    0    0    0    1    1   20    8    6    1   49   15
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    2    1
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    1    0    0    0    0    1    1    0    0    0    0    0    0    0    0    0
    0    0    0    1    2    3    6    0    2    0    1    0    0    1    0    0
    7    8   10    6   19   22   22   10    5    9    9    7   16   21    6   10
   13   21   19   20   48   47   19    7    8   20   11   18   34   34   29   19
   35   35   44   51   86   42    9    2    9   19   70   32   26   28   37   40
   13   10   11   12   53    4    0    3    2   17   62   45   14    9   31   42
    0    0    0    0    1    0    0    0    0    0    4    3    2    0   13    3
    0    0    0    0    2    0    0    0    0    0    0    0    0    0    1    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    1    0    0    0    0    0    0    2    1    0    1    1    1    1    0
    3    1    1    1    2    4    5    6    3    0    3    4    8    5    1    2
   15    6   14   19   19   28   25   13    5    9   11   17   19   15   17   20
   29   33   29   27   45   39   19    7    6   14   17   11   23   38   28   20
   57   32   40   57   83   36    9    4    5   21   61   43   29   26   45   35
   20   15   23   13   62    0    0    2    5   13   62   43   16   12   36   52
    0    0    0    1    6    0    0    0    0    0    2    8    3    1    8    7
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    6    4    3    2    2    5    8    4    4    5    1    3    4    3    4    3
    3    6    7    6    5   12   16   12    8   10   15   10   12    6    6    9
   21   21   39   23   27   33   59   36   27   28   33   36   37   40   22   18
  105   94   89   96   89  115  128   56   51   61   93   99  107   91   87  116
  167  164  149  114  178  161   93   33   44   82  103  102  104   97  109  160
  440  277  257  248  435  122   64   26   67  139  318  258  112   99  206  421
  184  111  104  116  427    7    9   19   30  119  263  201   72   45  279  472
    1    3    5   13   53    0    0    2    1    0   15   15    2    5   72   52
    0    0    0    0    5    0    0    0    0    0    0    1    0    0    4    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
```

TABLE 3. (Cont'd) PAVAN Input File - Aux. Bldg. West Release
(1994-98 meteorological data)

9	5	11	9	19	31	56	52	24	14	17	22	14	12	10	10
21	14	19	14	19	40	74	54	43	29	33	42	20	18	11	14
61	44	57	48	51	154	187	134	100	80	94	93	55	21	28	53
199	113	163	104	118	357	280	220	176	191	222	263	123	66	77	151
212	75	151	125	158	224	171	111	157	253	366	330	151	51	83	225
191	103	81	102	242	105	121	61	149	388	783	489	190	87	219	375
58	24	15	18	147	4	15	13	50	157	213	180	62	40	155	160
0	1	11	9	19	1	0	0	1	0	7	10	2	3	14	10
0	0	0	1	9	1	0	0	0	0	0	0	0	0	0	0
0	0	0	0	3	0	0	0	0	0	0	0	0	0	0	0
11	19	28	27	40	68	128	100	58	32	31	24	36	27	22	20
17	20	18	16	24	55	149	78	59	50	32	37	32	26	20	28
52	34	33	24	31	128	167	110	93	99	91	67	53	37	32	38
51	24	24	8	35	73	73	49	79	143	155	167	53	24	47	64
7	2	5	1	18	9	8	6	6	58	119	99	26	9	15	13
1	1	0	1	3	0	2	0	0	8	56	31	7	0	8	5
1	0	0	0	0	0	0	0	0	0	1	3	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
160	74	84	74	154	166	254	249	261	194	198	179	253	332	377	206
33	13	11	13	18	44	106	63	91	87	90	78	83	77	61	39
31	12	3	11	15	25	36	23	47	145	155	75	59	40	38	38
10	1	1	3	2	5	2	1	9	82	134	69	5	6	13	6
1	0	0	1	0	1	0	0	1	1	14	10	1	3	0	1
0	0	0	1	0	0	0	0	0	0	0	0	1	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
-1.	0.50	0.75	1.0	1.5	2.0	3.0	5.0	7.0	10.0	13.0					
503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.
4793.	4793.	4793.	4793.	4793.	4793.	4793.	4793.	4793.	4793.	4793.	4793.	4793.	4793.	4793.	4793.

TABLE 3. (Cont'd) PAVAN Input File - Aux. Bldg. East Release
(1994-98 meteorological data)

```
0100 00001
Surry (VA Power)      1994-98 JFD          Ground-Level Release
  9.6 m               9.6 m - 44.9 m
Onsite
Aux Bldg East Louver Release
   10    0
 960. 35.5  10.  9.6
    0    0    0    0    0    0    0
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    1    1    0    0    1    0    0    0    0
    0    1    3    0    1    2    1    3    1    1    0    0    0    2    0    0
    9    7    8   18   23   34   50   26   23   17    7   13   32   39   18    4
   41   42   65   95  163  186  158   35   35   49   38   43  135  174   95   46
  131  173  170  197  352  379  159   32   41  128  260  157  229  407  371  149
   64   89   58   95  319   10    3    0    6   69  370  156   98  208  300  201
    0    0    3    2   24    0    0    0    1    1   20    8    6    1   49   15
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    2    1
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    1    0    0    0    0    1    1    0    0    0    0    0    0    0    0    0
    0    0    0    1    2    3    6    0    2    0    1    0    0    1    0    0
    7    8   10    6   19   22   22   10    5    9    9    7   16   21    6   10
   13   21   19   20   48   47   19    7    8   20   11   18   34   34   29   19
   35   35   44   51   86   42    9    2    9   19   70   32   26   28   37   40
   13   10   11   12   53    4    0    3    2   17   62   45   14    9   31   42
    0    0    0    0    1    0    0    0    0    0    4    3    2    0   13    3
    0    0    0    0    2    0    0    0    0    0    0    0    0    0    1    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    0    1    0    0    0    0    0    0    2    1    0    1    1    1    1    0
    3    1    1    1    2    4    5    6    3    0    3    4    8    5    1    2
   15    6   14   19   19   28   25   13    5    9   11   17   19   15   17   20
   29   33   29   27   45   39   19    7    6   14   17   11   23   38   28   20
   57   32   40   57   83   36    9    4    5   21   61   43   29   26   45   35
   20   15   23   13   62    0    0    2    5   13   62   43   16   12   36   52
    0    0    0    1    6    0    0    0    0    0    2    8    3    1    8    7
    0    0    0    0    1    0    0    0    0    0    0    0    0    0    0    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
    6    4    3    2    2    5    8    4    4    5    1    3    4    3    4    3
    3    6    7    6    5   12   16   12    8   10   15   10   12    6    6    9
   21   21   39   23   27   33   59   36   27   28   33   36   37   40   22   18
  105   94   89   96   89  115  128   56   51   61   93   99  107   91   87  116
  167  164  149  114  178  161   93   33   44   82  103  102  104   97  109  160
  440  277  257  248  435  122   64   26   67  139  318  258  112   99  206  421
  184  111  104  116  427    7    9   19   30  119  263  201   72   45  279  472
    1    3    5   13   53    0    0    2    1    0   15   15    2    5   72   52
    0    0    0    0    5    0    0    0    0    0    0    1    0    0    4    0
    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0    0
```

TABLE 3. (Cont'd) PAVAN Input File - Aux. Bldg. East Release
(1994-98 meteorological data)

9	5	11	9	19	31	56	52	24	14	17	22	14	12	10	10
21	14	19	14	19	40	74	54	43	29	33	42	20	18	11	14
61	44	57	48	51	154	187	134	100	80	94	93	55	21	28	53
199	113	163	104	118	357	280	220	176	191	222	263	123	66	77	151
212	75	151	125	158	224	171	111	157	253	366	330	151	51	83	225
191	103	81	102	242	105	121	61	149	388	783	489	190	87	219	375
58	24	15	18	147	4	15	13	50	157	213	180	62	40	155	160
0	1	11	9	19	1	0	0	1	0	7	10	2	3	14	10
0	0	0	1	9	1	0	0	0	0	0	0	0	0	0	0
0	0	0	0	3	0	0	0	0	0	0	0	0	0	0	0
11	19	28	27	40	68	128	100	58	32	31	24	36	27	22	20
17	20	18	16	24	55	149	78	59	50	32	37	32	26	20	28
52	34	33	24	31	128	167	110	93	99	91	67	53	37	32	38
51	24	24	8	35	73	73	49	79	143	155	167	53	24	47	64
7	2	5	1	18	9	8	6	6	58	119	99	26	9	15	13
1	1	0	1	3	0	2	0	0	8	56	31	7	0	8	5
1	0	0	0	0	0	0	0	0	0	1	3	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
160	74	84	74	154	166	254	249	261	194	198	179	253	332	377	206
33	13	11	13	18	44	106	63	91	87	90	78	83	77	61	39
31	12	3	11	15	25	36	23	47	145	155	75	59	40	38	38
10	1	1	3	2	5	2	1	9	82	134	69	5	6	13	6
1	0	0	1	0	1	0	0	1	1	14	10	1	3	0	1
0	0	0	1	0	0	0	0	0	0	0	0	1	0	0	1
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
-1.	0.50	0.75	1.0	1.5	2.0	3.0	5.0	7.0	10.0	13.0					
503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.
4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.	4765.