VIRGINIA ELECTRIC AND POWER COMPANY
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Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
CONTROL ROOM HABITABILITY
OPERATOR DOSE ASSESSMENT

In conjunction with an engineering evaluation and dose assessment completed for control room habitability at North Anna Power Station, similar evaluations were performed for Surry Power Station. This assessment considered the radiological consequences of several other accident scenarios in addition to LOCA (i. e., waste gas decay tank and volume control tank ruptures, secondary pipe rupture, steam generator tube rupture, and fuel handling accident).

The original licensing basis analysis assumed that radiological consequences for the control room dose calculations would be the most limiting during the Design Basis Accident (LOCA). However, the recent assessment has shown that the other accidents evaluated may be more limiting for specific components of the control room dose (i. e., thyroid, whole body gamma, skin).

As noted in the attached report it was determined that our 1982 control room dose calculations for the Design Basis Accident (LOCA) remain valid. However, as a result of expanding the assessment to consider the additional accident scenarios, the limiting control room doses have increased above previously reported values. Therefore, an unreviewed safety question exists as defined by 10 CFR 50.59. In each case evaluated the predicted control room dose remains within the limits of 10 CFR 50, Appendix A, General Design Criterion 19.

Should you have any questions about this evaluation, please call.

Very truly yours,

W. L. Stewart
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Attachment

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Safety Evaluation
Surry Control Room Habitability

1.0 INTRODUCTION

During accidents at the Surry Power Station, the control room envelope is pressurized (upon receipt of a control room isolation signal) to 0.05 inch water gauge to minimize in-leakage of air and airborne radioactive material. The pressurization is initially provided by a bottled air system, which has sufficient capacity to supply the control room for 60 minutes. When the bottled air supply is depleted, breathing and pressurization air is provided by an emergency filtered air system which draws air from the turbine building through HEPA and charcoal filters. The emergency ventilation system maintains the control room envelope at a minimum pressure of 0.05 inch water gauge relative to the outside atmosphere.

Previously, radiological dose calculations for the control room during accidents, discussed in Section 11.3.6 of the Surry UFSAR, were performed only for the Loss Of Coolant Accident (LOCA), on the assumption that this Design Basis Accident (DBA) would be the most limiting case for calculation of control room doses.

Section 11.3.6 of the original Surry FSAR notes that the dose to control room personnel as the result of a LOCA does not exceed 2.5 rem whole body and 10 rem to the thyroid in 31 days. In 1982, revised control room doses were calculated and submitted to the NRC (Reference 1). These

calculations, which remain valid, assumed a 10 cfm unfiltered in-leakage to account for the impact of personnel access to the control room on the control room pressurization, and also incorporated site specific meteorological data. These doses were also determined for a range of emergency ventilation system air flow rates. For a 2200 cfm flow rate, the calculated doses were:

23.0 Rem Thyroid

0.14 Rem Whole Body Gamma

1.4 Rem Beta Skin

By comparison, Part 6.4 of the Standard Review Plan (NUREG-0800) and GDC-19 of 10 CFR 50 Appendix A define the following maximum permissible doses to control room personnel:

30 Rem Thyroid

5 Rem Whole Body Gamma

30 Rem Beta Skin

More recently, the question has been raised whether it is valid to assume that the LOCA is the limiting accident from a control room habitability perspective: other accidents might be more limiting due to proximity of the ventilation system intakes to the release sites for other accidents, or due to delays in isolating the control room (e.g., for a tank rupture, whereas the control room is automatically isolated on SI). Therefore, for the current reevaluation of control room habitability, doses to

control room personnel have been determined for the following additional accidents from Chapter 14 of the Surry UFSAR:

(1) Waste gas decay tank rupture

(2) Volume control tank rupture

(3) Major secondary pipe rupture

(4) Steam generator tube rupture

(5) Fuel handling accidents

Although no changes have occurred which would require reevaluation of the LOCA control room doses, the LOCA doses were also recalculated as part of the current study to ensure consistency. These evaluations incorporated site meteorological data through 1987 (which resulted in lower atmospheric dispersion factors, or X/Qs, and thus lower doses). The calculations were also conservatively based on an uprated power of 2554 MWt. This assumption will increase the inventory of short-lived isotopes available for release from failed fuel, and will conservatively increase the calculated consequences of any accidents.

The only Condition IV accident discussed in Chapter 14 of the Surry UFSAR which was not specifically addressed in the control room habitability calculations was the Rod Control Cluster Assembly (RCCA) ejection accident, for which it has been determined that less than 10% of the fuel will fail (Section 14.3.3 of the Surry UFSAR). The consequences of the limited amount of fission product release for the rod ejection accident are bounded by the analysis for the LOCA.

The Locked Rotor Accident (LRA) was also not evaluated for impact on the Surry control room doses. Although fuel failures are currently assumed to occur due to this accident, under the Surry design basis for this accident (locked rotor with coincident loss of offsite power) the radioactive releases are contained within the primary system. As there is no pathway for the activity from the fuel to be released to the environment other than normal leakage, the consequences of this accident are also bounded by the LOCA for Surry.

2.0 SYSTEM DESCRIPTION

The Surry control room habitability system for radiological protection includes a compressed breathing air system and an emergency filtered air system. The calculations described in Section 3.0 were performed to verify that operation of these systems under projected accident conditions will ensure doses to personnel within the control room do not exceed the limits of GDC-19.

For normal operations, the control and relay room area exhaust and replenishment supply ventilation (2500 cfm) is provided by external systems. Two cross-connected air cooling systems are provided per control room (for a total of four) to ensure cooling at all times, during both normal and accident conditions. Each air cooling system consists of an air handler and a chiller supply, and has sufficient capacity for the total cooling load.

In the event of an emergency, the normal outside air supply and exhaust are automatically isolated. The control and relay room area is sealed with weatherstripped doors, and by redundant seismic Category I isolation dampers on the normal supply and exhaust vents.

A compressed breathing air system is provided to maintain a positive control room pressure to assure outward leakage when the outside air is contaminated. The compressed breathing air system consists of two banks of air bottles. Each bank is designed to provide 18,000 cubic feet of free air, and will provide one hour of positive pressure. The control

room dose calculations assume that this system is automatically initiated upon receipt of a control room isolation signal, such as is generated by the Safety Injection System (SIS). (The bottled air supply will also be activated when the isolation dampers are closed manually.) As most accident analyses do not take credit for transit time necessary for the radioactive releases to reach the control room, automatic initiation of the bottled air system is generally required to ensure that the limits of GDC-19 are satisfied. Virginia Electric and Power Company is currently making the necessary modifications to provide this automatic initiation capability.

An emergency filtered air system is provided to ensure continued outward leakage and to continue the supply of breathing and pressurization air indefinitely upon depletion of the bottled air supply. Four 100%-capacity fans, each with a a rated capacity of 1000 cfm, are installed in parallel to provide emergency ventilation. One fan is used to supply emergency ventilation to each of the main control rooms and each of the emergency switchgear rooms. This system takes suction from the turbine building through roughing, particulate, and charcoal filters.

Two parallel, safety related charcoal filter assemblies are provided for emergency safeguards ventilation. These charcoal adsorbers reduce the potential release of radioactive iodine from the auxiliary building to the environment, which in turn reduces potential intake of radioactive iodine from the environment into the control room. The impact of these filters on the control room doses is incorporated into the appropriate evaluations, such as for the LOCA and fuel handling accidents.

Technical Specifications 3.23, 4.12 and 4.20 require that the charcoal filters have a radioactive methyl iodide removal efficiency of at least 95 percent for expected accident conditions. Per Technical Specification 3.23, the control room dose calculations assume only 90 percent iodine removal efficiency for air passing through the charcoal filters.

High efficiency particulate air (HEPA) filters are provided for removal of particulates from the emergency ventilation air intake. Surry Technical Specifications 3.23, 4.12, and 4.20 require that the HEPA filters have an efficiency of at least 99.5% removal of DOP particulates. Roughing filters installed before the HEPA filters and charcoal adsorbers prevent clogging of the iodine adsorbers.

Multiple entries to the pressurized control and relay room area are required by the emergency operating procedures. To account for the effects of this access to the control room during an accident condition, the control room dose calculations assumed a continuous unfiltered in-leakage of 10 cfm. This value was taken from Part 6.4 of the Standard Review Plan (NUREG-0800).

3.0 REVIEW OF CALCULATIONS AND ASSUMPTIONS

The accidents defined in Chapter 14 of the Surry UFSAR are those which result in the most significant impact on control room habitability. Specifically, the following accident conditions were addressed in this evaluation:

(1) Loss of Coolant (LOCA)

(2) Waste Gas Decay Tank (WGDT) Rupture

(3) Volume Control Tank (VCT) Rupture

(4) Main Steam Line Breaks (MSLB)

(5) Steam Generator Tube Rupture (SGTR)

(6) Fuel Handling Accident (FHA)

Calculations were prepared to determine the 30 day dose to control room inhabitants. These calculations were based on models and methodology which are consistent with NUREG-0800.

The input assumptions and results for each of these accidents are summarized below. In general, the analyses considered the automatic isolation of the control room and activation of the bottled air supply upon receipt of the SI signal. The exceptions are the two tank rupture accidents, for which the control room was not assumed to be isolated, and the fuel handling accident, for which manual control room isolation is assumed. (The reasons for these exceptions will be included in the discussions for these accidents.) Normal ventilation was assumed prior to isolation, and the time of isolation actuation was accident dependent.

After depletion of the bottled air system (60 minutes after the control room was isolated), pressurization and breathing air were provided by the emergency ventilation system for the remainder of each accident scenario. Consistent with Part 6.4 of the Standard Review Plan (NUREG-0800), an unfiltered in-leakage at 10 cfm was assumed for the duration of the accident to account for control room access.

1. Loss of Coolant Accident (LOCA)

The LOCA evaluation was based on methodology defined in Regulatory Guide 1.4 and NUREG-0800. Analysis of the consequences of this accident assumed that releases start at the moment the Safety Injection (SI) signal is initiated. The control room is automatically isolated, and the bottled air system is actuated, on an SI (that is, at T=0.0 hours). Transport time to the control room ventilation system intakes was conservatively treated as negligible (transport time=0.0 hours). The evaluation considered releases from the containment to the atmosphere for a duration of one hour, by which time the containment pressure will be reduced to subatmospheric, consistent with the Surry containment design basis analysis. The bottled air system provides positive pressure for 1 hour. At the end of this time, a filtered air intake of 1000 cfm to 4000 cfm is manually initiated (with the flow rate depending on the number of control and relay room emergency ventilation fans activated). The following filter efficiencies were used for this filtered air intake:

a. 90% removal of elemental iodines

b. 0% removal of methyl iodines

c. 90% removal of particulates

A 0% removal efficiency was also used for noble gases. The doses were calculated based on a safeguards area iodine filter efficiency of 90 percent. As noted previously, a 10 cfm unfiltered in-leakage was incorporated into the analysis for the duration of the accident (starting at T=0.0 hours) to account for the effects of multiple entries of the control room envelope during the accident.

A nominal core power level of 2554 MWt was used. A 2% instrument error was added, per Regulatory Guide 1.49, which resulted in a core power of 2605 MWt being used for the calculations. As discussed previously, this reflects a power uprating from the current nominal core power level of 2441 MWt.

The doses received due to ECCS leakage (over the entire 30 day period), and due to leakage from the containment (for the first hour of the accident, until the containment pressure is reduced to subatmospheric), were calculated. The dose due to direct shine from the containment was unchanged by the assumptions of the current evaluation, and was taken from a previous calculation of Surry control room doses. A conservative dose due to sky shine (i.e., shine from the containment which is not radiated directly toward the control room but which is redirected by scattering effects of the atmosphere) was also added. The total cumulative doses received in the control room

at the end of 30 days assuming a safeguards area filter efficiency of 90% iodine removal were determined to be:

	Emergency Ventilation		Allowable limits*
	1000 cfm	4000 cfm	
Thyroid:	26.6 rem	20.9 rem	30 rem
Whole Body Gamma:	0.51 rem	0.53 rem	5 rem
Beta (Skin):	1.25 rem	1.42 rem	30 rem

The results do not differ significantly from the doses reported to the NRC in 1982 (Reference 1):

	Emergency Ventilation		Allowable limits
	2200 cfm	4000 cfm	
Thyroid:	23.0 rem	22.8 rem	30 rem
Whole Body Gamma:	0.14 rem	0.192 rem	5 rem
Beta (Skin):	1.4 rem	1.65 rem	30 rem

As noted previously, no changes have occurred which required reanalysis of the Surry control room doses for a LOCA. However, to ensure consistency with the recent evaluations of the other accident scenarios, the LOCA was also reevaluated. The major differences from the previous analysis reviewed by the NRC which were incorporated into the most recent LOCA calculation are: the use of updated X/Q values (which actually decrease the doses); the assumed uprated power level

* As prescribed in SRP 6.4, and based on GDC-19, Appendix A, 10CFR50.

of 2554 MWt (nominal), which increases the doses; and the inclusion of the sky shine contribution to the whole body gamma doses. The most recent control room doses for the LOCA have increased slightly from the previous submittal, but will continue to be within the allowable limits.

2. Waste Gas Decay Tank (WGDT) Rupture

The analysis of the waste gas decay tank rupture was based on Branch Technical Procedure ETSB 11-5 (Part 11.3 of the Standard Review Plan, NUREG-0800).

There are two WGDTs, which are operated independently from each other and are separated by closed valves. The waste gas cycle for each tank typically consists of 30 days of feed, followed by 20 days of decay and 10 days of bleed.

The maximum activity in the WGDT is at the end of the feed cycle with the recombiner operating. The WGDT concentrations given in Table 11.2-3 of the current Surry UFSAR were used. These activities are based on operation of two units at 2546 MWt with 1% failed fuel, and assume that 100% of the gases and 1% of the iodine are removed from the letdown flow in the gas stripper (and subsequently routed to the waste gas decay tanks). A 17 gpm total letdown flow rate to the gas stripper from both units was used. The values from the UFSAR were then normalized to the limit of 24,600 Ci of Xe-133 equivalent which is given in Technical Specification 3.11.B.6. Only release of noble

gases was considered in this evaluation. Although some iodine may be present in the tank, the amounts are orders of magnitude below those considered for other accidents. Therefore, the thyroid dose for the WGDT rupture will be bounded by other accident conditions, and was not explicitly determined in this evaluation.

The tank burst was assumed to result in a "puff" release. Because of the short duration of the release and the lack of radiation monitors in the vicinity of the tank, the control room is not isolated for this accident. Therefore, the normal control room ventilation rate of 2500 cfm was used for this analysis.

The cumulative doses received by personnel in the control room at the end of 30 days due following a WGDT rupture were calculated to be:

		Allowable limits
Thyroid:	(Not calc.)	30 rem
Whole Body Gamma:	0.505 rem	5 rem
Beta (Skin):	19.7 rem	30 rem

These whole body and skin doses exceed the values previously reported for the LOCA, but are still within the limits of 10 CFR 50, Appendix A, GDC-19.

3. Volume Control Tank (VCT) Rupture

The analysis of the volume control tank rupture was performed in a manner similar to the waste gas decay tank rupture evaluation.

The entire volume control tank noble gas inventory, based on the concentrations currently given in Table 9.1-6 of the Surry UFSAR and the vapor volume given in UFSAR Table 9.1-5, is "puff" released upon rupture of the VCT. These activities are based on operation at 2546 MWt with 1% failed fuel, but were scaled to give the corresponding activities for 2605 MWt (2554 MWt nominal uprated core power plus 2% instrumentation uncertainty). Only the release of noble gases was considered in this evaluation. Therefore, the analysis for the VCT rupture only determined the whole body gamma and beta skin doses (i.e., no thyroid dose).

Because of the short duration of the release and the ineffectiveness of filtration systems for removing noble gases (which are the primary source of radioactivity for this type of release), it was assumed that the control room is not isolated for this accident. Therefore, the normal control room ventilation rate of 2500 cfm was used for this analysis.

The cumulative doses received by personnel in the control room at the end of 30 days due to a VCT rupture were calculated to be:

		Allowable limits
Thyroid:	(Not calc.)	30 rem
Whole Body Gamma:	0.594 rem	5 rem
Beta (Skin):	24.2 rem	30 rem

These whole body and skin doses exceed the values previously reported for the LOCA, but are still within the limits of 10 CFR 50, Appendix A, GDC-19.

4. Main Steam Line Break (MSLB)

The analysis of the main steam line break accident was based on guidelines provided in parts 6.4 and 15.1.5 of the Standard Review Plan, NUREG-0800. The break was assumed to be a double ended rupture, occurring outside containment in the turbine building. The MSLB analysis also considers a coincident loss of offsite power, which means that the main condenser is not available for cooldown.

The entire liquid inventory in the affected steam generator (including the primary liquid which leaks to the secondary side) is released in the initial 30 minutes. The entire secondary side gas inventory of the affected steam generator is released in a single "puff" at the initiation of the break. The affected steam generator is isolated after 30 minutes. The remaining releases considered in this analysis are due to the primary to secondary leakage in the unaffected steam generators. Releases occur from the unaffected steam generators (through the main steam relief valves) for 8 hours after the MSLB occurs.

The primary coolant and secondary side concentrations used for the analysis were based on the plant Technical Specifications. All iodines in the affected steam generator were conservatively assumed

to be released to the environment (PF = 1.0). For the unaffected steam generators, only 1% of the iodines which may be present on the secondary side (due to primary-to-secondary system leakage) were released to the atmosphere (PF = 0.01). The iodine partition factor for the unaffected steam generators is consistent with the value used previously for analysis of the MSLB for North Anna, and is given in part 15.1.5 of the Standard Review Plan.

As for the LOCA analysis, for the MSLB analysis the reactor trip and initiation of releases both occur on generation of the SI actuation signal, at T=0.0 hours. The transport time of the releases to the control room ventilation intakes was treated as negligible (T=0.0 hours). The SI signal automatically initiates the isolation of the control room and activation of the bottled air system. The bottled air pressurizes the control room for 1 hour, after which time a filtered air intake of 1000 cfm is available. A control room filter efficiency of 90% removal of elemental iodines was assumed, in accordance with the requirements of Surry Technical Specification 3.23, with all released iodine being in an elemental form. Once again, a 10 cfm unfiltered in-leakage was used for the duration of the accident, starting at T=0.0 hours, to allow for the effects of multiple entries to the control room envelope during the accident.

Doses were calculated for two cases: (1) a pre-accident iodine spike to 10 times the Technical Specification limit, and (2) a concurrent (accident initiated) iodine spike, which lasts for 4 hours. The calculated doses included releases from both the affected and

unaffected steam generators. No (additional) fuel failures were assumed to occur due to the MSLB accident, consistent with the design basis given in Chapter 14 of the Surry UFSAR.

The cumulative doses received by personnel in the control room at the end of 30 days were calculated to be:

	Pre-accident Iodine Spike	Concurrent Iodine Spike	Allowable Limits
Thyroid:	1.43 rem	1.68 rem	30 rem
Whole Body Gamma:	2.24E-4 rem	2.73E-4 rem	5 rem
Beta (Skin):	9.03E-3 rem	9.48E-3 rem	30 rem

The calculated control room doses for the MSLB accident are bounded by the results of the LOCA analysis, and are well within the allowable limits based on 10CFR50 Appendix A, GDC-19.

5. Steam Generator Tube Rupture (SGTR)

Although the Westinghouse Owners Group is still continuing its effort to define the most appropriate way to model this accident, an evaluation of the SGTR event is necessary to thoroughly evaluate control room habitability. Therefore, a conservative basis was selected for this control room dose calculation for the steam generator tube rupture accident. Once the new methodology is finalized, the Surry control room doses for the SGTR will be reassessed to determine the impact.

The SGTR analysis followed the methodology presented in part 15.6.3 of the Standard Review Plan, NUREG-0800. The break was assumed to be a double ended rupture, occurring near the top of the steam generator tube bundle, so that the break may be exposed if the liquid level in the steam generator drops below the top of the steam generator tube bundle. The defective steam generator was assumed to be isolated within 30 minutes of the accident, which is consistent with previous analyses of the SGTR, as discussed in Section 14.3.1 of the Surry UFSAR. It was also conservatively assumed that the main condenser was not to be available for steam dump due to coincident loss of offsite power. The two unaffected steam generators were therefore vented using the main steam relief valves for a period of 8 hours. Prior to the SGTR, the core was assumed to be operating at a power of 2605 MWt (2554 MWt nominal uprated power plus 2% instrument uncertainty).

The primary and secondary side concentrations used for this analysis were based on the plant Technical Specifications. The releases to the atmosphere start when the secondary side pressure exceeds the PORV setpoint at T=4 minutes. Once again, transport time from the site of the releases to the control room ventilation intakes was considered to be negligible (transport time=0.0 minutes). The steam generator tube bundle remains covered until the reactor trip, at 4 minutes after the accident. The tubes were then uncovered for 14 minutes. The duration of tube uncovery was determined in a manner consistent with that used to determine the tube bundle uncovery period for the most recently approved SGTR analysis for North Anna. For consistency with

the approved SGTR analysis, 1% of the iodine was released from the steam generators (PF=0.01) for the unaffected steam generators, as well as for the affected steam generator whenever the tube bundle was covered. During the period of tube uncovery, the iodine partition factor was conservatively set to PF=1.0 for releases from the affected steam generator.

During a SGTR, the control room is isolated and the bottled air system is actuated automatically on an SI signal. This occurs 5.0 minutes after the tube actually ruptures. As significant releases to the atmosphere do not start until 4 minutes after the tube ruptures, the total length of time after releases start during which only unfiltered air enters the control room is only 1.0 minute. The bottled air supplies the control room for 1 hour, after which a filtered air intake of 1000 to 4000 cfm is available (depending on the number of control and relay room emergency ventilation fans which are activated by the operators). The control room filters were assumed to be 90% effective for the removal of iodines, in accordance with Surry Technical Specification 3.23. All iodines were considered to be elemental for this evaluation. A 10 cfm unfiltered in-leakage was used for both the control room isolation period and when the emergency ventilation system was in use, to account for multiple entries of the pressurized control room envelope during the accident.

Doses were calculated for two cases: (1) a pre-accident iodine spike to 10 times the Technical Specification activity limit, and (2) a concurrent (accident initiated) iodine spike, which lasted for 4

hours. No additional fuel failures were assumed to occur due to the SGTR accident.

For the pre-accident iodine spike case, the cumulative doses received by personnel in the control room at the end of 30 days were calculated to be:

Emergency Ventilation:	1000 cfm	4000 cfm	Allowable Limits
Thyroid:	16.2 rem	7.01 rem	30 rem
Whole Body Gamma:	7.4E-3 rem	3.8E-3 rem	5 rem
Beta (Skin):	0.295 rem	0.14 rem	30 rem

For the concurrent iodine spike case, the 30-day doses received by control room personnel were determined to be:

Emergency Ventilation:	1000 cfm	4000 cfm	Allowable Limits
Thyroid:	1.81 rem	0.824 rem	30 rem
Whole Body Gamma:	6.7E-3 rem	3.4E-3 rem	5 rem
Beta (Skin):	0.290 rem	0.132 rem	30 rem

The calculated control room doses for all Surry SGTR cases are bounded by the results of the LOCA analysis, and clearly fall within the allowable limits based on 10CFR50 Appendix A, GDC-19.

6. Fuel Handling Accident (FHA)

The evaluation of the fuel handling accident assessed the cumulative doses received by control room personnel for 30 days following a fuel handling accident in the spent fuel pool. Analysis of the fuel handling accident was based on guidance provided in parts 6.4 and 15.7.4 of the Standard Review Plan (NUREG-0800) and in Regulatory Guide 1.25 (Safety Guide 25).

For evaluation of this accident, the fuel was assumed to have operated at a core power level of 2605 MWt (nominal core power of 2554 MWt plus a 2% instrument error), and the radial peaking factor for the damaged fuel assembly was assumed to have been 1.65. (This is higher than the 1.55 currently allowed by the Technical Specifications, and will result in a conservatively high inventory in the fuel.)* The fuel assembly with the greatest amount of activity (the highest power assembly) is damaged during the FHA. The FHA occurs 100 hours after reactor shutdown, which is the earliest time at which the fuel can be moved, per Surry Technical Specification 3.10.

Per Regulatory Guide 1.25, the gap activity in the damaged fuel assembly is assumed to consist of 10% of the iodines, 10% of the noble gases (except Kr-85) and 30% of the Kr-85 in the assembly. One hundred percent of the gap activity is released to the spent fuel pool

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* Note that this represents a conservative change from the basis of Technical Specification 3.10, which states that the FHA was analyzed based on damage to an assembly which had operated at a core average power of 2550 MWt.

during the FHA (i.e., all fuel rods in the assembly are damaged). The minimum depth of the water between the top of the damaged fuel rods and the fuel pool surface was 23 feet. All noble gases released from the damaged fuel assembly were released to the atmosphere, but only 1% of the iodines released from the assembly were released from the pool (PF = 0.01). The iodines which are released to the atmosphere consist entirely of elemental iodines, and the releases continue for 2 hours. The activity is released from the fuel building through the iodine filters, since fuel building ventilation exhaust is diverted through the charcoal filters whenever refueling is in progress. An iodine removal efficiency of 90% was considered for these filters, which is consistent with the control room iodine filter efficiency required by the Surry Technical Specifications.

Prior to the fuel handling accident, the normal ventilation system was in operation. The control room is then manually isolated by the control room operators before the radioactive release from the fuel building reaches the control room normal ventilation intake. This assumption is based on operator reaction to high radiation alarms in the fuel building (specifically, the fuel pool bridge area monitor). The bottled air system is also manually actuated by an operator (effectively at T=0.0 minutes for the control room personnel). The bottled air system provides envelope pressurization for 1 hour, after which time a filtered air intake of 1000 cfm is available. A 10 cfm unfiltered in-leakage was used during the control room isolation period (starting at T=0.0 hours) and when the emergency ventilation

system was in use, to account for multiple entries of the control room envelope during the accident.

The calculated 30-day cumulative doses to control room personnel following a fuel handling accident were:

			Allowable limits
Thyroid:	0.919	rem	30 rem
Whole Body Gamma:	1.87E-3	rem	5 rem
Beta (Skin):	0.120	rem	30 rem

These doses are well within the limits of 10 CFR 50, Appendix A, GDC-19, and are also bounded by the results for the LOCA.

The results of the dose calculations are summarized in Table 1. The most limiting cases are compared with the LOCA results last submitted to the NRC in Table 2.

Table 1

30-Day Doses In The Control Room

Accident	Thyroid Dose (Rem)	Gamma Dose (Rem)	Beta Dose (Rem)
LOCA - uprated			
1000 cfm intake	26.6	0.51	1.25
4000 cfm intake	20.9	0.53	1.42
MSLB with pre-accident iodine spike	1.43	2.24 E-4	9.03 E-3
MSLB with concurrent iodine spike	1.68	2.73 E-4	9.48 E-2
FHA	0.919	1.87 E-3	0.120
SGTR with pre-accident iodine spike	16.2	7.4 E-3	0.295
SGTR with concurrent iodine spike	1.81	6.7 E-3	0.290
VCT Rupture	-	0.594	24.2
WGDT Rupture	-	0.505	19.7
Allowable Limits	30.0	5.0	30.0

Table 2

Comparison of Limiting Cases
with Previously Submitted LOCA Doses

Accident	Thyroid Dose (Rem)	Gamma Dose (Rem)	Beta Dose (Rem)
LOCA - uprated			
1000 cfm intake	26.6	0.51	1.25
4000 cfm intake	20.9	0.53	1.42
VCT Rupture	-	0.594	24.2
WGDT Rupture	-	0.505	19.7
LOCA - 1982 submittal			
1500 cfm intake	24.5	0.12 *	1.4
2200 cfm intake	23.0	0.14 *	1.4
4000 cfm intake	22.8	0.192*	1.65

* Note that these doses did not include the conservative skyshine contribution included in the most recent whole body gamma doses.

4.0 SUMMARY AND CONCLUSIONS

The doses to Surry control room personnel during accident scenarios have been reevaluated to assess the impacts of accidents other than the LOCA on control room habitability. Although reevaluation of the LOCA was not necessary, this accident was also analyzed to ensure that the most recent evaluations of all accidents were on a consistent basis. The calculations included a 10 cfm unfiltered in-leakage to account for multiple access to the control room pressure envelope during accidents. To ensure that the limits of GDC-19 are not exceeded, the bottled air supply must be automatically initiated upon SI. These dose calculations were based on this premise.

All accident scenarios that can have a significant impact on control room operator doses were evaluated and determined to meet the limits specified by GDC-19. The LOCA is the most limiting accident for the thyroid doses. For the beta (skin) and whole body gamma doses, the volume control tank rupture is the most limiting case, followed by the waste gas decay tank rupture. The doses for these cases are high because the control room is assumed not to be isolated following the tank rupture accidents. The most bounding accident for the beta (skin) and whole body gamma doses for which the control room is isolated is the LOCA.

5.0 10 CFR 50.59 EVALUATION

This safety evaluation has described the effects of assessing accident scenarios other than the LOCA on the control room dose calculations for Surry. It is necessary for the control room to be automatically isolated and the bottled air system to be automatically initiated on receipt of an SI signal to mitigate the effects of contaminated air entering the control room. The LOCA continues to be the limiting case for determining the dose to the thyroid. However, other accidents have been found to give more limiting results for the whole body gamma and beta skin doses. Therefore, the control room dose consequences have increased from those reported to the NRC in 1982, which were based solely on the LOCA. It is concluded that an unreviewed safety question exists as defined in 10 CFR 50.59, and consequently the results are being submitted to the NRC for approval. The results of this evaluation can be stated as follows:

1. No increase in the probability of occurrence of an accident will occur. There are no changes which increase the probability of any given accident occurring. The evaluation of accident scenarios other than the LOCA results in predicted control room doses which increase from the doses most recently reported to the NRC. Therefore, an unreviewed safety question technically exists as defined in 10 CFR 50.59. Although the predicted consequences of an accident will increase for control room personnel, the allowable dose limits defined in GDC-19 of Appendix A of 10 CFR 50 are still met for each accident.

2. A more comprehensive evaluation of the impact of the existing UFSAR accidents on control room habitability was performed, but no new accident types or equipment malfunction scenarios were introduced. There is no possibility of occurrence of an accident of a different type than any previously evaluated in the UFSAR.

3. The margin of safety is not reduced. An evaluation of the accidents which have the greatest impact on control room dose calculations was performed. It has been concluded that the acceptance criteria defined by GDC-19 will be met.

6.0 BASIS FOR NO SIGNIFICANT HAZARDS DETERMINATION

It has also been determined that the reevaluation of the Surry control room doses does not involve a significant hazards consideration as described in 10 CFR 50.92. The results of this determination can be stated as follows:

1. There is no significant change in the probability or consequences of an accident previously evaluated. There are no system changes which increase the probability of an accident occurring. The effects on the analysis for each accident have been investigated, and the doses to control room personnel were found to increase. This increase is not significant because the revised doses remain below the limits in GDC-19 of Appendix A of 10 CFR 50, and meet the guidelines of NUREG-0800 (Section 6.4).

2. No new accident types or equipment malfunction scenarios have been introduced. Therefore, the possibility of an accident of a different type than any evaluated previously in the UFSAR is not created.

3. There is no reduction in the margin of safety. The revised dose calculations for all accidents continue to meet the appropriate GDC-19 limits.

7.0 REFERENCE

1. Letter from R. H. Leasburg (Virginia Electric and Power Company) to Harold R. Denton (NRC), "Auxiliary Ventilation System Modification, Proposed Technical Specifications, Surry Power Station - Units 1 and 2," Serial Number 335, June 7, 1982.