Control Room Radiological Assessment

Dresden and Quad Cities Stations

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Commonwealth Edison Company

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Introduction

The nuclear power plant control room (CR) dose acceptance limits are established by 10 CFR 50, Appendix A General Design Criteria (GDC) Number 19. The requirements of GDC-19 became a design basis for Dresden and Quad Cities Stations as a result of the commitments made to NUREG 0737, Item III.D.3.4. This post-TMI action item required a review of the CR ventilation system, which included an assessment of the system's ability to adequately protect the CR operators against the effects of accidental release of radioactive gases and verification that the plant can be safely operated and shut down under design basis conditions. Commonwealth Edison (ComEd) made commitments in the responses to NUREG 0737, Item III.D.3.4, to upgrade the CR ventilation systems for Dresden and Quad Cities to comply with the 30-day post-accident dose limits. These upgrades to the Control Room Emergency Filtration System (CREFS) included the installation of a new air handling unit (AHU) train with an air filtration unit (AFU) at both Dresden and Quad Cities. The resulting systems at both plants included a safety-related (Train B) AHU in addition to the original non-safety-related (Train A) AHU. A charcoal filter unit was installed to provide post-accident filtered air to either AHU to minimize the introduction of activity to the CR emergency zone following a design basis accident. These upgrades were reviewed and approved by the NRC (References 1 & 2).

Subsequently, the analyses were revised to change two major inputs. The first input was the assumption that only 10 cfm of unfiltered air entered the CR emergency zone. As a result of an airborne event at Zion, (Licensee Event Report 86-035 dated September 11, 1986), walkdowns of the Dresden and Quad Cities CREFS were performed. The walkdowns revealed that the CR emergency zone would leak in excess of the 10 cfm assumed in the original analyses. This infiltration is in the form of inleakage through the ductwork and components under a negative pressure located outside of the CR emergency zone. This leakage was calculated to be 263 cfm for Dresden and 260 cfm for Quad Cities. The second input to be corrected was the use of SBGTS methyl iodide removal efficiency of 99%. This original input was based on the FSAR, whereas the plant Technical Specification value was 90%. Following the correction of these original inputs, the design basis analyses calculated a 30-day post LOCA CR operator Thyroid dose of 29.3 rem and 29.4 rem for Dresden and Quad Cities, respectively.

The CR radiological analysis utilized at the stations were determined to require evaluation during the fall of 1996. This was based on the identification of nonconforming conditions. CREFS was not maintaining the entire emergency zone at the desired positive pressure, the Secondary Containment volume was smaller than described in the UFSAR and Technical Specifications, and the unfiltered infiltration was potentially in excess of the analytical value used to calculate the CR radiological consequences of a design basis accident. While the nonconforming conditions have been corrected at both stations, ComEd determined that it would be prudent to reexamine the issues related to the CR dose analysis and update the existing analyses. This report incorporates current industry methods and guidance which in turn demonstrates a substantial increase in operating margin between the calculated consequences of design basis accidents and the GDC-19 acceptance limit.

While the primary objective of the Topical Report is to refine the radiological design basis of the Control Room, the resulting release path model for a loss-of-coolant accident is also being used to refine the radiological consequence analysis for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ).

The most limiting accident was determined to be the design basis loss-of-coolant accident (LOCA) in the previous radiological dose analyses. ComEd has reviewed other design basis accidents that could potentially affect CR dose as part of this effort. This assessment of other accidents was performed for both Dresden and Quad Cities using the conservative NUREG-0800 (reference 3) approach as opposed to the methodologies currently described in the UFSARs. The analysis revealed that the main steam line break (MSLB) (for ground level type releases) and the LOCA (for elevated releases) constitute the most-limiting scenarios with respect to CR operator dose.

Report Organization

This report is arranged into four sections. The first section of the report is background information and a description of the Dresden and Quad Cities Control Room Habitability Systems. The second section of the report is a description of the revised dose analysis methodology, assumptions, inputs, and conservatism's. The third section of the report is a summary of the results and the final section provides the conclusions.

In addition, appendices, figures, and tables are included which provide results of the revised analyses.

Section I

Background Information and A Description of The Dresden and Quad Cities Control Room Habitability Systems.

Background Information and a Description of the Dresden and Quad Cities Control Room Habitability Systems

Background

The radiological consequences of design basis accidents were assessed during the original Dresden and Quad Cities licensing process using many of the conservative methodologies employed by General Electric during that time frame. As part of the licensing process, the NRC performed a number of conservative confirmatory analyses which demonstrated the adequacy of the facility. These NRC SER analyses (which were prepared prior to the issuance of NUREG-0800) did not fully define the analysis assumptions or methodology that were employed and as such could not be adopted by ComEd as the radiological design basis.

The CR dose acceptance limits are established by 10 CFR 50 Appendix A, GDC-19. The requirements of GDC-19 became a design basis for Dresden and Quad Cities Stations as a result of the commitments made to NUREG 0737, Item III.D.3.4. This post-TMI action item required a review of the CR ventilation system, which included an assessment of the system's ability to adequately protect the CR operators against the effects of accidental release of radioactive gases and verification that the plant can be safely operated and shut down under design basis conditions. ComEd also made commitments to upgrade the CR ventilation systems for Dresden and Quad Cities to comply with the 30-day post-accident dose limits. These upgrades to the Control Room Emergency Filtration System (CREFS) included the installation of a new air handling unit (AHU) train, including a Refrigeration Control Unit, along with an air filtration unit (AFU) at both Dresden and Quad Cities. The resulting systems at both plants included a safety-related (Train B) AHU to serve as the primary train during an accident and the original non-safety-related (Train A) AHU became the backup. The AFU was installed to provide post-accident filtered air to either AHU to minimize the introduction of activity to the CR emergency zone following a design basis accident.

Subsequently, two design inputs to the original analyses were revised. The first input was the assumption that only 10 cfm of unfiltered air entered the CR emergency zone. As a result of an airborne event at Zion in 1986, walkdowns of the Dresden and Quad Cities CREFS were performed. The walkdowns revealed that the CR emergency zone unfiltered inleakage would be in excess of the 10 cfm assumed in the original analyses. This infiltration is in the form of inleakage through the ductwork and components under a negative pressure located outside of the CR emergency zone. This inleakage was calculated to be 263 scfm for Dresden and 260 scfm for Quad Cities. The second input to be corrected was the use of an SBGTS methyl iodide removal efficiency of 99%. This original input was based on the FSAR, where as the plant Technical Specification value was 90%.

The reanalysis of the post-accident operator dose, with these revised input parameters, resulted in a 30-day thyroid dose in excess of the GDC-19 limits. The operation of the CREFS at both stations was revised so that the dose was within GDC-19 acceptance limits. The original design for each station was based on allowing the system to supply unfiltered air for the first 8 hours of the accident. The dose was reduced to within acceptance limits by requiring initiation of the CREFS within 40 minutes after an accident for Dresden and within 110 minutes for Quad Cities. These changes were considered consistent with the original dose submittals, since the assumed manual response times were in accordance with the guidance document (SRP 6.4) used to review the submittals. SRP 6.4 states that "A substantial time delay should be assumed where manual isolation is assumed, e.g., 20 minutes for the purpose of dose calculations." The 40 minute and 110 minute manual isolation times are longer than the time suggested by SRP 6.4 as "substantial". The NRC accepted these changes as a result of their inspection (Reference 4).

ComEd agreed with the NRC staff in late 1996 and early 1997 to review the accident analyses to confirm which scenario(s) produced the most-limiting results with respect to CR operator dose and to include those scenarios in this report. Other accident scenarios were reviewed to ensure a more limiting condition does not exist. This review revealed the need to quantitatively analyze the MSLB outside containment, in addition to the LOCA for CR dose impact.

ComEd discovered in late 1996 and early 1997 that the Dresden and Quad Cities Secondary Containment volumes were actually less than that documented in the UFSAR and Technical Specifications. The NRC issued an amendment to Dresden Units 2 and 3 to revise the Technical Specifications to specify SBGTS filter methyl iodide removal efficiency of 95% (Reference 5) thereby offsetting the effect of the smaller secondary containment volume. A similar amendment was approved for Quad Cities Units 1 and 2 (Reference 6). This efficiency is credited in the current and revised analyses for Dresden and Quad Cities and is confirmed periodically by testing in accordance with Technical Specifications.

Recently, a number of BWR Owners Group member plants have requested and received revisions to their Technical Specifications allowance for higher Main Steam Isolation Valve (MSIV) bypass leakage. Those requests were based, in part, on credit for suppression pool scrubbing in accordance with SRP 6.5.5. Dresden and Quad Cities Stations are not requesting a Technical Specification change for MSIV leakage at this time, however, we have included the impact of higher MSIV leakage on CR dose in this report.

Control Room Habitability System Description

General

The control room habitability systems for Dresden and Quad Cities Stations consist of systems and equipment which protect CR operators against postulated releases of radioactive materials, toxic gases, and smoke by isolation and pressurization of the CR emergency zone. Emphasis in the following descriptions is placed on the habitability systems for the Control Rooms relative to demonstrating radiological protection.

Although there are design differences, both Dresden and Quad Cities use a similar approach to protecting CR operators during plant emergencies. The Control Rooms at both stations are part of a defined emergency zone within which operators can safely remain while they carry out their emergency duties. The Control Room emergency zones at both stations have two separate heating, ventilation and air conditioning trains (AHU Trains A & B) that supply outside air which is mixed with recirculated air. Train A normally supplies air during normal plant operations.

The CREFS, which contains the AFU, supplies 2000 ± 10% scfm outside air to the emergency zone through a prefilter, electric heater, high efficiency particulate air (HEPA) filters, two 2-inch activated charcoal adsorber banks, and a post charcoal HEPA filter. The system is designed to remove radioactive iodine and particulate matter from incoming air during a radiological emergency. The habitability systems at both stations provide operator radiological protection in a manner that is best described as zone isolation with filtered pressurization (Emergency Mode).

Dresden Station Control Room Habitability System

The following signals automatically isolate the CR emergency zone and close the dampers for both trains A and B:

- Toxic gas detection

- Smoke detection in the outside air intake (Train A only)

CR radiological protection is accomplished by manually (within 40 minutes) closing the normal outside air intake to the operating Air Handling Unit (either Train A or B). At the same time, the AFU is activated to pressurize the CR emergency zone by supplying it with filtered outside air.

The CR emergency zone allows the CR to be maintained as the center from which emergency teams can safely conduct operations during a design basis radiological release. Within this emergency zone the CR operators are adequately protected against the effects of accidental radioactive releases. This emergency zone for

Dresden Station Units 2 and 3 includes the main CR, the kitchen, the wash room, the locker room, the auxiliary computer room, and the Train B HVAC equipment rooms.

Dresden Station plans to remove the auxiliary computer room from the CR emergency zone. Removal of the auxiliary room from the emergency zone reduces the potential for infiltration of contaminated air via leakage into return ducts.

The boundaries of the Control Room emergency zone envelope are shown in UFSAR Figure 6.4-1, "Dresden Control Room HVAC Schematic", along with a simplified diagram of the CR HVAC system. UFSAR Figure 6.4-2, "Control Room Arrangement", shows the arrangement of equipment in the CR and the points of entry. This figure also shows the wall which was erected to separate the Unit 1 CR area from the Units 2 and 3 area. Note that the Train A equipment room is located on the level immediately above the CR, over the Train B equipment room. The auxiliary computer room is located on the level immediately below the CR. UFSAR Figure 6.4-3, "General Plant Layout", is a view of the general plant layout, showing the location of radioactive material release points and CR air inlets.

The CREFS supplies 2000 ± 10% scfm of outside air to either operating AHU to maintain the areas of the CR emergency zone at a positive pressure of at least 1/8 inch water gauge with respect to adjacent areas during the emergency mode. The AFU conforms with Regulatory Guide 1.52 and is located within the Train B HVAC equipment room inside the CR emergency zone.

HVAC Train B is a single zone system which supplies necessary cooling in the event of an emergency, or upon failure of Train A. The air distribution from each train is aligned through the use of air-operated isolation dampers. The Train A dampers fail to the Train B mode since this train is powered from the emergency bus during a loss of offsite power (LOOP).

The CR HVAC system has three operating modes in addition to normal operation:

* The emergency mode protects CR personnel from airborne radioactive contaminants. In this mode the normal outside air intake is isolated, and the AFU provides filtered outside air makeup to pressurize the CR emergency zone.

* The isolation/recirculation mode protects personnel from toxic gases. In this mode all outside air intakes are isolated, and CR air is recirculated.

* The smoke/isolation purge mode (Train A only) protects personnel from smoke. CR air is recirculated with the outside air intake and purge exhaust dampers isolated when smoke is sensed in the outside air intake. When smoke is sensed in the return, the outside air intake and purge exhaust dampers are opened to purge smoke from the space.

The following HVAC system operating parameters pertain to the current CR dose calculations:

* Unfiltered outside air is drawn into the CR at a rate of 2000 ± 10% scfm until the CR is isolated and pressurized by operator action within 40 minutes after the occurrence of a LOCA. Unfiltered inleakage of 263 scfm is being drawn into the CR.

* 263 scfm infiltration of contaminated air from areas outside the CR emergency zone enters the CR emergency zone after the CR is isolated and pressurized.

Quad Cities Station Control Room Habitability System

The following signals automatically isolate the CR emergency zone and close the normal intake dampers for both Trains A and B:

* High radiation in either the drywell, Reactor Building, or Refuel Floor
* High main steamline flow
* Low reactor water level
* Toxic gas detection
* Smoke Detection in outside Air Intake (Train A only)

The operating train upon isolation will continue to recirculate air to provide cooling for the CR emergency zone. The AFU is aligned manually to the operating train to pressurize the CR emergency zone to at least 1/8 inch water gauge with respect to adjacent areas by supplying it with filtered outside air (emergency mode) within the specified time, currently set at 60 minutes in the procedures and 110 minutes in the analyses.

Quad Cities Units 1 and 2 CR emergency zone includes the CR, the cable spreading room, the auxiliary equipment room, the old computer room, and the Train B HVAC equipment room. However, the Train B HVAC equipment room is excluded from the CR emergency zone when Train A is operating because it is not required to protect against unfiltered inleakage. This zone adequately protects operators from the effects of accidental radioactive gas releases and provides a center from which emergency teams can manage emergency operations. The Train A HVAC equipment room is outside the CR emergency zone. Support rooms such as the kitchen, offices, and washrooms are accessible to the operators during an emergency with the aid of breathing equipment.

The boundaries of the CR emergency zone envelope are shown in UFSAR Figure 6.4-1, "Quad Cities Control Room HVAC Schematic", which includes a simplified schematic of the CR HVAC system. UFSAR Figure 6.4-3, "Quad Cities Control Room Layout", shows the arrangement of equipment in the CR and its points of entry. Although it is within the CR emergency zone, the Train B HVAC equipment room is located on the mezzanine floor outside the CR proper and at a slightly lower elevation. UFSAR Figure 6.4-4, "Quad Cities Station Control Room Habitability General Plant Layout", shows dimensions, location of radioactive material release points, and location of CR air inlets.

The CREFS supplies 2000 ± 10% scfm of outside air to either operating AHU to maintain the areas of the CR emergency zone at a positive pressure of at least 1/8 inch water gauge with respect to adjacent areas during the emergency mode. The AFU conforms with Regulatory Guide 1.52 and is located adjacent to the Train B HVAC equipment room outside the CR emergency zone.

HVAC Train B is a single zone system which supplies necessary cooling in the event of an emergency, or upon failure of Train A. The air distribution from each train is aligned through the use of air-operated isolation dampers. The Train A dampers fail to the Train B mode since Train B is powered from the emergency bus during a Loss Of Offsite Power (LOOP).

The CR HVAC system has the following operating modes in addition to normal operation:

* The emergency mode protects CR personnel from airborne radioactive contaminants. In this mode the normal outside air is automatically isolated on the above signals and the AFU provides filtered outside air to pressurize the CR emergency zone.

* The isolation/recirculation mode is used to protect personnel from toxic gases. In this mode all outside air intakes are isolated and CR air is recirculated.

* The smoke isolation/purge mode (Train A only) protects personnel from smoke. with the outside air intake and purge exhaust dampers isolated when smoke is sensed in the outside air intake. When smoke is sensed in the return, the outside air intake and purge exhaust dampers are opened to purge smoke from the space.

The following HVAC system operating parameters pertain to the current CR dose calculation:

* The HVAC system air intake dampers are automatically closed in the event of a LOCA and HVAC system goes into the recirculation mode. The current analysis conservatively considers that unfiltered air enters the CR emergency zone at a rate of 1260 scfm until the AFU has been manually started.

* The AFU is required to be manually started within 110 minutes in the current analysis even though the current operating procedures require 60 minutes.

Section II

Methodology, Assumptions, Inputs and Conservatism Dresden and Quad Cities Loss of Coolant Accident

Methodology, Assumptions, Inputs and Conservatism

Dresden and Quad Cities Loss of Coolant Accident

The existing licensing basis dose analysis has been examined and compared to NRC licensing guidance (primarily Reg. Guides 1.3, and SRP 6.4, 6.5.3, 6.5.5, 15.6.5, 15.6.5 Appendix A, 15.6.5 Appendix B) to confirm the acceptability of the analysis input parameters. The existing input parameters also have been checked for technical viability and the extent to which they conform with NRC regulatory requirements and guidance. The supporting revised analyses are consistent with the previous CR dose analyses submitted in response to NUREG 0737, Item III.D.3.4. Changes have been made to the existing input parameters to more appropriately conform with NRC guidance and to utilize current industry methodologies to predict accident doses. The proposed changes involve the following:

* added credit for suppression pool scrubbing
* included credit for International Commission on Radiological Protection ICRP-30 (Reference 7) dose conversion factors
* reduced amount of mixing in the reactor building from 100% to 50%
* revised CR emergency zone
* reduced CREFS manual initiation time from 110 minutes to 40 minutes (Quad Cities only)
* added dose contribution from Engineered Safeguard Feature (ESF) leakage in secondary containment
* utilized recent calculated (i.e., reduced) reactor building volume
* added flow tolerance and margin to CREFS filter unit flow and SBGTS flow
* revised the MSIV extrapolation factor to a more conservative value
* assessed impact of increased MSIV leakage
* assessed impact of source term for Siemens high burnup fuel
* assessed impact of higher CR unfiltered leakage
* assessed secondary containment bypass leakage

The most significant of these are credit for suppression pool scrubbing in accordance with SRP 6.5.5, use of the ICRP 30 dose conversion factors, and the reduction of the credit for mixing in the reactor building from 100% of the volume to 50%. The cumulative effect of adding all of the flow tolerances and margins also adds a sizable degree of conservatism.

Assumptions and input parameter values for the CR dose calculations for Dresden and Quad Cities are summarized in Table 1. The analysis performed to assess the radiological consequences of a Design Basis LOCA was performed following the general guidance contained in Regulatory Guides and Standard Review Plan. Table 1 identifies the current licensing basis parameter, the revised parameter and the basis for the parameter. The new methodology incorporates several less restrictive assumptions along with a number of more restrictive assumptions. The net affect, however is a new calculation methodology that is still very conservative. Table 1 identifies some of the conservatisms in the revised analysis. Key assumptions and input parameter values are described below, along with a discussion of the principal quantitative and/or qualitative conservatisms, as appropriate. These conservatisms are in addition to the extremely conservative methodology developed in TID-14844 (reference 8) to assess the adequacy of reactor sites.

AXIDENT Code Input

Regulatory Guide 1.3 and TID-14844 have been used to determine activity levels in the containment following a design basis LOCA. The "AXIDENT" computer code was used to calculate CR dose. This program utilizes a time dependent model which considers the rate of change of activity based on the inleakage, exhaust, cleanup, decay, etc. This AXIDENT methodology provides a more realistic evaluation than the steady-state Murphy-Campe model. The AXIDENT code is the current licensing basis methodology.

NUREG/CR 5659 published in December 1990, provides a transient analysis computer program to assess the control room habitability in lieu of the steady state Iodide Protection Factor (IPF) presented by Murphy/Campe. NUREG/CR 5659 also incorporates more realistic methodology for atmospheric dispersion factors based on the work at Pacific Northwest Laboratory by Ramsdell. The revised analysis does not utilize the less conservative atmospheric dispersion factors of the most recent methodologies.

The CR inhalation dose analysis consists of six separate AXIDENT runs. For each release pathway (i.e., SBGTS effluent, ESF Leakage, and MSIV bypass leakage), a CR calculation is performed for each of the two distinct "infiltration periods" (0 to 40 minutes and 40 minutes to 720 hours). This revised analysis focuses on the operator dose as a result of the inhalation of activity and the immersion in the cloud of activity within the CR. The direct shine whole body dose from the activity within the facility and the shine whole body dose from the release plumes are based on the analysis work performed for the original habitability submittal. The dose contribution from these mechanisms are small with respect to the acceptance criteria.

Atmospheric Dispersion Factor (X/Q)

The design basis atmospheric dispersion factors (X/Q) for the CREFS intake are based on the previous habitability submittals. The previous design basis atmospheric dispersion factors for the stack releases were based on the conservative values in Reg. Guide 1.3. The ground level X/Q's were based on the Halitsky methodology (Reference 9) as opposed to the Murphy/Campe methodology (Reference 9) recommended by SRP 6.4. The previous habitability submittal utilized the general Murphy/Campe equation for ground level dispersion (i.e., X/Q = Ka/AU) except that a Ka value of 2 was used as opposed to the Murphy/Campe expression (Ka = K + 2). The value of Kc = 2 was chosen for the previous analysis based on review of the available test results. The Murphy/Campe equation was based on dispersion around circular PWR type containment's whereas the Haltsky data was based on square edged buildings. A detailed discussion of the basis for using the methodology was provided in the previous licensing submittals. This discussion concluded that "sufficient data and field tests exist to provide reasonable assurance that the chosen X/Q is conservative, over and above the conservatism implied by using the fifth percentile wind speed and wind direction factors." Based on the degree of conservatisms in the previous submittals, the atmospheric dispersion factors were not revised for this revised analysis.

StandBy Gas Treatment System Filter Efficiency

A SBGTS charcoal filter methyl iodide removal efficiency of 95% is used for the revised analysis, consistent with the recent technical specification changes approved for Dresden (Reference 5) and for Quad Cities (Reference 6). This efficiency is conservatively applied to all forms of iodine even though the actual removal of particulate and elemental iodine's by the charcoal adsorber is expected to be above 95%.

Secondary Containment Bypass Leakage

Both stations were reviewed for the potential of additional bypass leakage paths. This review entailed identifying all potential paths which originate in the containment or are attached to a system which penetrates the containment and that ultimately terminates or passes through an area outside of the secondary containment. Each path was reviewed in a realistic manner to determine if a potential bypass leakage path exists. The revised analysis does not include any bypass leakage paths other than MSIV leakage (see appendices B and C).

Reactor Building Zone Volume

The reactor building volumes (Reference 5 and 6) are used in the revised analyses.

Control Room Emergency Zone Volume

The CR emergency zone volume is 81,000 cu.ft. for Dresden and 184,000 cu.ft. for Quad Cities. Based on the planned removal, the volume of the auxiliary computer room is excluded from the Dresden CR emergency zone. The Quad Cities Train B HVAC Equipment Room is not considered part of the CR emergency zone during the operation of HVAC Train A. Smaller CR volumes have the effect of raising the initial concentration of radionuclides for a given unfiltered intake rate (time = 0 to 40 minutes). This effect is offset by the faster purging of radionuclides for a given filtered intake flow (time = 40 minutes to 30 days). Sensitivity studies were performed which showed that the calculated 30-day thyroid dose is insignificantly affected by the volume of the zone.

Control Room Intake

The activity which enters the main CR may be the result of MSIV leakage, SBGTS exhaust or both, depending on wind direction. As a result of the location of these sources with respect to the CREFS intake, it is possible for the intake to be exposed to activity from both sources simultaneously. The analyses conservatively assumes that the activity concentration at the intake is due to concurrent MSIV leakage and stack releases for the duration of the event.

Control Room Infiltration

The infiltration of unfiltered air into the CR emergency zone utilizes three paths: (1) through the CR emergency zone boundary; (2) through the system components located outside of the CR emergency zone; and (3) through the backflow at the CR emergency zone boundary doors as a result of personnel ingress or egress.

Infiltration through the CR emergency zone is assumed to be zero when the CR ventilation system is pressurized based on the guidance of Standard Review Plan 6.4. The CREFS supplies 2000 ± based on 10% scfm of outdoor air to maintain a minimum positive CR pressure of 1/8 inch water gauge relative to adjacent contaminated areas during the emergency mode. This positive pressure differential prevents infiltration through the CR emergency zone by ensuring that air is exfiltrating from the CR emergency zone.

The current design basis unfiltered inleakage values of 263 scfm and 260 scfm for Dresden and Quad Cities respectively, were derived from calculations performed in the 1987 time frame. These values were recently verified through actual testing. The as-left measured inleakage at both Dresden and Quad Cities are less than the calculated inleakage. The inleakage was measured using tracer gas tests. The tracer gas tests were performed using a procedure which is based on ASTM Standard E741-93, "Standard Test Method for

Determining Air Change Rate in a Single Zone by Means of a Tracer Gas Dilution." The tests were accomplished using an electronegative gas, sulfur hexafluoride (SF6), as a tracer. This gas was utilized since it is recognized as non toxic, non reactive, inert, and easily detectable in minute quantities by means of electron capture gas chromatography.

The opening and closing of CR emergency zone doors can induce infiltration to the CR emergency zone. Backflow infiltration is conservatively assumed at 10 scfm, as recommended by Standard Review Plan 6.4, which is consistent with the current analysis.

Primary Containment Leakage

The enclosed analysis specify a primary containment leakage rate of 1.6% per day for Dresden and 1.0% per day for Quad Cities. These are the containment leak rates specified in the Technical Specifications for the stations and those used in the current CR dose studies. These primary containment leakage's represent the "total" leakage (i.e., the sum of both the MSIV leakage plus the primary to secondary leakage). The existing primary to secondary containment leakage is held constant in the analysis that assesses the impact of higher allowed MSIV leakage.

MSIV Leakage

The treatment of MSIV leakage in the previous dose analyses is being maintained for this revised analysis. As in the previous CR dose analysis, credit is taken for iodine plateout on the surfaces of the main steam lines and turbine-condenser complex. Realistically components of the main steam lines and the turbine-condenser complex would remain intact following a design basis LOCA. The previous dose methodology utilized conservative deposition velocities. The resulting plate out is at least as conservative as those previously approved by the NRC. Discussions from the previous submittals along with a discussion of the modeling refinements follow.

MSIV leakage bypasses both the primary and secondary containment by way of the steam lines. A total MSIV bypass leakage rate of 46 scfh for all lines (11.5 scfh per MSIV at 25 psig per Technical Specifications) has been used in the analyses, as well as higher total leakage values, as shown in Table 1. The leakage rates were corrected to the containment design pressure, using the laminar (viscous) flow extrapolation factors of ORNL NSIC-5 (Reference 10). Radioactivity leaking past the isolation valves could be released through the outboard MSIV stems into the steam tunnel or continue down the steam lines to the stop valves and into the turbine-condenser complex. Leakage into the steam tunnel is exhausted by the SBGTS filtration system, thus eliminating it as a bypass pathway. For conservatism, no MSIV leakage was assumed to enter the steam tunnel.

Leakage down the steam lines is subject to plate out and decay within the lines. NUREG/CR-009, Section 5.1.2 discusses iodine removal rates which can be applied to calculate plateout on the piping and turbine-condenser surfaces. Elemental and particulate iodine decontamination factors of over 100 can be calculated for small travel distances and large travel times down the steam lines, considering the small volumes of leakage.

The MSIV leakage travels down the steam piping to the turbine-condenser complex where it is released as a ground-level release at a rate of 1% of the turbine-condenser volume per day. This leak rate is consistent with the guidance provided for the control rod drop accident in SRP 15.4.9.

The revised analysis conservatively treats the MSIV leakage path as a single volume (as opposed to three separate volumes) with a total surface area of 650,000 square feet and a total volume of 170,000 cubic feet. The MSIV leakage actually passes through three volumes, which provide holdup and the opportunity for plateout. The first volume consists of the steam lines between the inboard and outboard isolation valves, the second volume consists of the steam lines between the outboard isolation valve and the turbine stop valves, and the third volume includes the steam lines after the turbine stop valves and the turbine-condenser complex.

The previous analyses credited the additional delay associated with concentration buildup within the three separate volumes.

The iodine removal rates were calculated for elemental and particulate iodine, using a deposition velocity of 0.012 cm/sec. Removal of organic iodine through plateout is not considered.

The MSIV leakage will enter the turbine building, where it will be exhausted by the HVAC system, if in operation. Additional plateout on ductwork and fans will further minimize the iodine releases. Should the HVAC system not be in operation, MSIV leakage will tend to collect in the building and be subject to additional decay and plateout. However, once the MSIV leakage reaches the turbine building , the revised analysis conservatively assumes that no additional plateout or decay occurs.

Secondary Containment Mixing

Regulatory Guide 1.3 recommends that the analysis assumes that the primary containment leakage pass directly to the SBGTS without mixing in the secondary containment. The previous CR dose analyses takes credit for mixing in 100% of the secondary containment volume. Mixing with 50 percent of the secondary containment volume is assumed for the revised analysis, consistent with SRP 6.5.3.II. The leakage from the primary containment can not "short circuit" to the release point, hence the 50% mixing basis is justified based on the SBGTS configurations of both Dresden and Quad Cities.

Secondary Containment Release Rate

The secondary containment volume has been calculated to be smaller than that originally described in the UFSAR. The smaller volume effectively increases the release rate of activity to the environment. The radiological dose assessment calculations that form the basis for conformance to GDC-19, use an activity release rate from the secondary containment of 1 volume per day. The concentration and quantity of post-accident isotopes being released to the environment and to the CR atmosphere is increased with the reduced reactor building volume.

The assumed release rate is further increased by using the maximum allowable SBGTS flow (i.e., 4000 cfm plus 10%) in addition to adding 10 percent margin.

Control Room Intake Flow

The CR intake flows in the previous analyses were based on nominal design flows. The flows were revised to account for the allowable Technical Specification tolerance. Specifically , the CREFS flow during pressurization is assumed to be 1620 scfm (2000 scfm design flow less 10% Technical Specification tolerance less 10% additional margin). Prior to pressurization Dresden's revised analysis used intake flow of 2000 scfm + 10% and 263 scfm of unfiltered inleakage (i.e., 2463 scfm unfiltered in leakage for the first 40 minutes) because the intake is not automatically isolated. The current analysis for Quad Cities was based on automatic isolation of the CR unfiltered intake at time zero. The revised analysis conservatively assumes that the intake flow continues as if the intake would not have closed. Quad Cities revised analysis used 2000 scfm + 10% and 260 scfm of unfiltered in leakage (i.e., 2460 scfm unfiltered inleakage for the first 40 minutes). The Radiological Control Room models for Dresden and Quad Cities are provided in Figures 1 and 2, respectively.

Pressure Suppression Pool Scrubbing

Pressure suppression pool scrubbing with a minimum decontamination factor (DF) of 5 was applied to the amount of particulate and elemental iodine leaking from the primary containment to the secondary containment in accordance with SRP 6.5.5.III.1. The SRP notes that, for a Mark I containment, the applicant's decontamination factor of 5 or less may be accepted without any need to perform calculations A DF of one is used for organic iodides.

The initial airborne isotopic activity in the primary containment may be adjusted to account for the suppression pool scrubbing effect on iodine. Activity released from the core during the blowdown phase of a LOCA will be mixed in the drywell atmosphere. As a result of the pressure buildup in the drywell, the steam/air mixture will be forced through the downcomers into the suppression pool where condensables are removed. Iodine fission products are scrubbed in the process of passing through the suppression pool water. The scrubbing of iodine is limited to particulate and elemental iodine because organic iodine is more subject to dissolution. GE NEDO-25420 (Reference 11) presents pool DF data which justifies suppression pool scrubbing factors of 30 to 1000 for elemental and particulate iodine species. A large break LOCA with an instantaneous release of fission products postulated at the beginning of the event would result in most of the blowdown and activity passing rapidly through the suppression pool.

A slower, more mechanistic activity release would result in less activity being available instantaneously for release to the reactor enclosure. However, the slow release would be accompanied by steam and hydrogen (e.g., NUREG/CR-2540) which would pressurize the drywell and force flow through the suppression pool where significant quantities of iodine would still be removed. In addition, emergency cooling water circulation from the reactor to the drywell through the suppression pool and back to the core by core spray and LPCI would contribute to scrubbing of iodine released from the core long after blowdown. Furthermore, considering the conservative application of a DF of 5 to the primary containment leakage path, the net effect of both leak paths on the CR operator dose justifies the MSIV leak path scrubbing with a DF of 5. As a result, the application of the minimum decontamination factor of 5 to all release paths including the MSIV leak path, which may bypass the suppression pool but is still in contact with suppression pool water, is reasonable.

ICRP 30 DCFs

The existing licensing basis accident analysis is based on the DCFs from Regulatory Guide 1.3 and TID-14844, which were developed in the early 1960s. RG 1.109 recommends DCFs that are significantly lower than those specified in RG 1.3 or TID-14844 (ICRP-30 provides lower DCFs). Although these DCFs have not been included in a regulatory guide for use in accident analyses, they have been submitted to and approved by NRC in a number of post-TMI control room dose analyses. The ICRP-30 DCFs will be utilized in this methodology.

ESF Leakpath

The ESF CR dose contribution is subject to the guidance of SRP 15.6.5, Appendix B. Both stations implemented a program to reduce leakage from ESF components which circulate water outside of the primary containment as a result of NUREG-0737, post-TMI action Item III.D.1.1. This leakage is typically assessed in the industry in a realistic manner and is treated as an administrative limit which requires the plants to take action should the limit be exceeded. Dresden and Quad Cities procedures do not have an allowable "as left" leakage. The current programs ensure that the leakage is maintained at a very minimal value. Historically, the dose analyses conservatively use the typical industry administrative limit of 5 gallons per hour. The ESF leakrate is taken as two times the sum of the administrative limit for simultaneous leakage from all components in the recirculation systems (10 gals/hr) in the revised analysis.

The ESF CR dose contribution is modeled separately and added to the doses form the SBGTS and MSIV leakage. Fifty percent of core iodine inventory, based on maximum reactor power level, is assumed mixed in the pressure suppression pool water circulating through the containment external piping systems. Iodine is analyzed as being uniformly mixed in the minimum volume of the pressure suppression pool water in accordance with SRP 15.6.5 Appendix B. This volume is conservatively bounded by a value of 110,000 cu.ft. for Dresden and Quad Cities. The actual ESF water volume is substantially larger because it would include reactor water and water in piping.

The flash fraction is taken to be 10 percent in accordance with SRP 15.6.5 Appendix B since the temperature of the pressure suppression pool water circulating outside of containment does not exceed 212 degrees F (the

Pool Condensation Stability Limit is 204 degrees F for Dresden and 205 degrees F for Quad Cities). Ten percent of the iodine in the leakage is thus assumed to become airborne. The airborne activity released by flashing ESF water is assumed to mix with half the air in the secondary containment before it is released through the SBGTS to the environment via the stack.

Exfiltration From the Secondary Containment

The need to consider secondary containment exfiltration during the draw down-period was reviewed. The secondary containment building was designed to have a leakage no greater than 100% of the volume per day (which, based on the original volume calculations, was equal to 4000 cfm). During normal plant operation the reactor building is maintained at a vacuum of greater than or equal to 0.25 inches water gauge (Dresden) and 0.10 inches water gauge (Quad Cities). This is verified at least once every 24 hours in accordance with the Technical Specifications. The normal reactor building HVAC isolates and SBGTS auto starts during an accident. At least once per 18 months, secondary containment integrity is verified by operating one SBGTS subsystem at a flow rate of less than or equal to 4000 cfm for one hour and maintaining a vacuum greater than or equal to 0.25 inches of vacuum water gauge in the secondary containment in accordance with Technical Specifications.

All the activity which passes from the primary to the secondary containment is discharged to the environment via the SBGTS system. The analysis does not consider ground level exfiltration during either the draw-down period or during steady state operation.

Siemens Source Term Comparison

The current licensing basis utilizes TID-14844 source terms. The TID-14844 source terms are used as a basis for this revised analysis. In addition, a new source term for the 2-year cycle Siemens fuel (ATRIUM-9B fuel) was also evaluated in the revised analyses. Siemens source term information was provided in Reference 12. Siemens fuel source terms for 20,000 MWd/MTU and 60,000 MWd/MTU which is equivalent to a core residence time in days of 909 and 2727 were used. While the TID-14844 source terms with 1000 day burnup is considered the design basis, the revised analyses assess the effect of the Siemens specific source term at various burnups.

Dresden and Quad Cities Main Steam Line Break

The MSLB accident analysis assumptions and methodology used to assess the CR dose were based on accident analysis Reactor Coolant Release Data from Dresden and Quad Cities. UFSARs and a conservative transport methodology that meets Standard Review Plan 15.6.4. The impact of a MSLB on the CR dose was not presented in the original habitability study submitted to the NRC in response to NUREG 0737, Action Item III.D.3.4. The original submittal presented the LOCA as the limiting accident regarding CR dose.

The general MSLB accident analysis assumptions and methodology are based on the general guidance provided in SRP 15.6.4, SRP 2.3.4 and Reg Guide 1.5.

Two iodine concentrations are analyzed in accordance with SRP Section 15.6.4. These concentrations correlate technical specification values associated with (1) the maximum equilibrium value permitted for continued full power operations and (2) the maximum value permitted corresponding to an assumed pre-accident iodine spike. The quantity of liquid reactor coolant assumed to be released was based on the current UFSAR values which correspond to a MSIV closure time of 10.5 seconds. This is conservative compared to the Technical Specification MSIV closure time requirement of 3 to 5 seconds. The fraction of iodine which becomes airborne was assumed to be equal to the fraction of coolant flashing to steam in the depressurization process based on guidance in SRP 15.6.2. No credit is taken for dose mitigation, such as plateout, confinement in the Turbine Building, or decay during transit. Credit is take for decay and cleanup effect within the CR due to the introduction of clean air following the was conservatively modeled as a uniform sphere, with a uniform concentration, that passes over the CR intake at a rate of 1 meter per second, with no credit for atmospheric dispersion. The model also neglects the buoyancy of the steam cloud, conservatively assumes that the cloud stays at ground level and that the centerline of the cloud passes by the CR intake.

The analyses assumed that the only vehicle to introduce activity to the CR is through the normally open outside air intake which was modeled at a rate of 2000 scfm plus a 10% margin. No credit was taken for operation of the CREFS during the duration of the event. No credit was taken at Quad Cities for protection provided by the automatic isolation of outside air intake that would occur as a result of the high main steam line flow. The analysis excludes the introduction of activity to the CR from the infiltration through ductwork in adjacent areas since these areas will have activity concentration significantly less than the direct intake from the cloud. Operator exposure is based on ICRP 30 DCFs and a constant breathing rate.

Due to the Conservative Modeling Techniques employed (i.e., 10.5 seconds release, excluding the buoyancy of steam) and due to the low probability of a main steam line break coincident with a preaccident iodine spike, the resulting analysis is considered very conservative.

Modeling Approach for AXIDENT Code

The MSLB source term, as discussed previously, is treated as a constant flow of air with a uniform radionuclide concentration being drawn into the CR as the cloud passes, followed by a continuous flow of clean air at the same flow rate. This simplified CR source term model simulates a cloud of radioactive air with a uniform concentration of radionuclides, generated by a MSLB, passing over the CR air intake. The radionuclide concentration leaving the containment and entering the CR is then maintained in the specifying a X/Q of 1.0.

Dresden and Quad Cities LOCA Offsite Dose

Analyses performed to assess the offsite radiological consequences of a Design Basis LOCA using the proposed LOCA accident analysis methodology. The analyses were performed to determine the 2-hour Exclusion Area Boundary (EAB) dose and the 30-day Low Population Zone (LPZ) dose. The analyses were performed using the TID-14844 source terms and the atmospheric dispersion factors from Regulatory Guide 1.3 (fumigation was applied for the first 1/2 hour) in conjunction with the CR LOCA methodology described herein. The methodology used in the offsite dose analyses is the same as the CR dose analyses. The revised analyses evaluate offsite dose with 50% mixing in secondary containment and with no mixing in secondary containment.

Section III

Summary of Results

Summary of Results

The Dresden LOCA 30-day integrated CR doses for TID source terms are presented in Table 2. The results using Siemen's fuel at 20,000 MWd/MTU burnup and 60,000 MWd/MTU burnup source terms are presented in Table 2A. Corresponding results for Quad Cities are presented in Tables 3 and 3A, respectively.

The Dresden and Quad Cities MSLB CR doses are presented in Table 4.

The effect on Dresden CR operator thyroid dose as a function of infiltration for various infiltration values is shown in Figure 3, with MSIV leakage at the current Technical Specifications limits, and Figure 4, with total MSIV leakage of 120 scfh. Corresponding results for Quad Cities are presented in Figures 5 and 6.

Table 5 identifies the results of the offsite dose revised analyses. This table also identifies the original EAB and LPZ dose determined by the AEC and furnished in the SERs.

Section IV

Conclusions

Conclusions

ComEd has developed a revised methodology for calculating dose to the control room operators during postulated radiological events at Dresden and Quad Cities stations. The original control room habitability studies were provided in the early 1980's in responseto NUREG-0737, Item III.D.3.4. The revised methodology described herein significantly enhances the control room dose assessment by adopting updated radiological methodologies, revised dose conversion factors, suppression pool scrubbing effects, and higher burnup fuel.

The revised methodology continues to demonstrate adequate margin to regulatory limits as specified in GDC-19. For example, the revised control room radiological analysis for the Loss of Coolant Accident yields a 30-day thyroid dose of 9.93 rem for Dresden and 8.28 rem for Quad Cities compared with the original design basis exposures of 29.31 rem and 29.28 rem respectively (holding input assumptions constant).

Using a consistent approach, the offsite dose consequences were also evaluated for a postulated Loss of Coolant Accident and shown to be well below 10CFR 100 limits.

Commonwealth Edison requests NRC Approval of the this report, in as much as it licensing basis for the Control Room Radiological Analysis and the LOCA offsite radiological consequences.

Commonwealth Edison expects that future revisions to other design basis accident analyses for the Dresden and Quad Cities units will be accomplished, consistent with the provisions of 10 CFR 50.59, based on this analytical methodologies and assumptions. In addition, approval of these analyses may serve as part of the basis for a future request for less restrictive MSIV leakage Technical Specifications.

Appendix A - References

(1) Control Room Habitability Study for Dresden Units 2 and 3, Commonwealth Edison Company, Prepared by Bechtel Power Corp. Revised December 1981, submitted to the NRC via letter dated December 17, 1981, E. Douglas Swartz to D.G. Eisenhut.

(2) Control Room Habitability Study for Quad Cities Unit 1 and 2, Commonwealth Edison Company, Prepared by Bechtel Power Corp. Revised December 1981, submitted to the NRC via letter dated December 17, 1981, E. Douglas Swartz to D.G. Eisenhut.

(3) U.S. Nuclear Regulatory Commission Standard Review Plan (NUREG-0800), Sections:

2.3.4, "Short term dispersion estimates for accidental atmospheric releases."

6.4, "Control Room Habitability System."

6.5.3, "Fission Product Control Systems and Structures."

6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System."

15.4.9, "Spectrum of Rod Drop Accidents (BWR)."

15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside of Containment."

15.6.4, "Radiological Consequences of a Main Stream Line Failure Outside Containment."

15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Piping Breaks Within Reactor Coolant Pressure Boundary."

15.6.5, Appendix A, "Radiological Consequence of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution."

15.6.5, Appendix B, "Leakage From Engineered Safety Feature Components Outside Containment."

(4) Memorandum from J.A. Zwolinski to C.E. Norelius, dated 7/3/91, Region III's Request to Evaluate the Adequacy of the Quad Cities and Dresden As-Built Control Room Air Cleaning Systems to Meet the Requirements of 10 CFR Part 50, Appendix A, GDC-19.

(5) Letter dated April 25, 1997 from J. Stang, U.S. Nuclear Regulatory Commission, to I. Johnson, Commonwealth Edison Company, Issuance of Amendment (TAC Nos. M98389 and M98290).

(6) Letter dated March 27, 1997 from R.M. Pulsifer, U.S. Nuclear Regulatory Commission, to I. Johnson, Commonwealth Edison Company, Issuance of Amendments (TAC Nos. M98227 and M98228).

(7) ICRP Publication 30, "Limits for Intakes for Radionuclides by Workers."

(8) TID-14844 "Calculation or Distance Factors for Power and Test Reactor Sites, " March 23, 1962.

(9) Nuclear Power Plants Control Room Ventilation System Design For Meeting General Criterion 19, " K.G. Murphy and Dr. K.M. Campe, 13th AEC Air Cleaning Conference.

(10) ORNL NSIC-5, " U.S. Containment Technologies", August 1965.

(11) NEDO-25420, "Suppression Pool Scrubbing Factor for Postulated Boiling Water Reactor Accident Conditions", June 1981.

(12) Siemens letter JHR: 96:188, dated May 20, 1996 for the Quad Cities/Dresden Nuclear Power Stations.

Appendix B - Dresden Bypass Leakage Evaluation

Purpose

The purpose of this evaluation is to validate that the 46 scfh (at 25 psig) main steam isolation valve (MSIV) leakage is the bounding leakage value for all leakage paths for purposes of radiological calculation after a loss of coolant accident (LOCA) for Units 2 and 3. The criteria and methodology are listed below.

Background

Pathways from primary containment to points outside of secondary containment boundary which bypass secondary containment (and the Standby Gas Treatment System) exist at Dresden. With the exception of the MSIV leakage pathway, these potential bypass pathways have not been explicitly considered (assigned a separate leakage value) in the radiological assessments of LOCA scenarios. SRP Guidance for evaluating control room dose requires a review of all potential bypass pathways.

Criteria

All penetrations of primary containment were included in the evaluation. The following criteria and assumptions were used to evaluate if the lines need to be considered for secondary containment bypass leakage:

1. Potential leakage paths that terminate in the secondary containment are excluded.
2. All valves were assumed to leak, however, 3 or more closed valves in series would result in negligible leakage, allowing the line to be excluded.
3. Water-filled paths can be eliminated based on additional scrubbing in the water.
4. Lack of credible driving force 10 minutes after a LOCA (concern is for first 10 min.). Ref. UFSAR Fig. 6.2-19.
5. The Standby Gas Treatment System is operating.
6. The Emergency Core Cooling System pumps are operating.
7. The HRSS Ventilating System remains intact following a LOCA.

Basis

The basis for the evaluation methodology is from guidance provided by the NRC and is as follows:

1. The Standard Review Plan (SRP) 15.6.5, Appendix A states (middle of page 15.6.5-10) that the secondary containment bypass leakage rate must be considered. Also (page 15.6.5-10, bottom): "For dual containment systems, the bypass leakage is evaluated. This leakage, usually expressed as a fraction or percentage of the primary containment leak rate, is assumed to pass from primary containment directly to the environment, bypassing the secondary containment."

2. SRP 6.2.3 (pg. 6.2.3-5) states "The fraction of primary containment leakage bypassing secondary containment and escaping directly to the environment should be specified. Branch Technical Position (BTP) CSB 6-3 provides guidance ..."

3. BTP CSB 6-3 (pg. 6.2.3-11, item 6) states "The total leakage rate for all potential bypass leakage paths ... **should be determined in a realistic manner** ... This value should be used in calculating the

off-site radiological consequences of postulated loss of coolant accidents and in setting technical specification limits with margin for bypass leakage."

The NRC has not provided any specific guidance to quantify the "realistic manner" in which the bypass leakage is to be determined. A review of two cases is as follows: In 1976, a question was asked of license applicants to identify and quantify potential bypass leakage, LaSalle responded with a line list and arguments that multiple isolation valves, water filled legs of piping, et cetera were sufficient to preclude any bypass leakage in its design. This was accepted by the NRC. Fermi-2 responded similarly except that they stated conservatively, 4% of the containment leakage (0.5%/day) would bypass secondary containment; the NRC accepted this and included it in their SER assessment.

The HRSS Ventilation System is assumed to remain intact for several reasons. First, non-seismic ducting/piping does not fail in a seismic event due to its inherent ruggedness. The ducting/piping may deform, but no loss of pressure boundary occurs. Second, Loss of Offsite Power (LOOP) will reduce airflow through the HRSS Ventilation System. The HRSS Ventilation System fans lose power during a LOOP, but air will still flow through the system since air will still be flowing out the main plant vent stack - the 'chimney effect' will draw some air through. Third, a tornado that could completely disable the HRSS Ventilation System is not postulated to occur simultaneously with a LOCA.

Methodology

The evaluation considered all system piping routes connected to primary containment that also penetrate the secondary containment boundary. Further details on credibility of individual leakage path potential are shown in the comments of Table II.

The methodology includes the following sources of conservatism:

- In many cases, the criterion of single active failure stated in NUREG-0800 (7/81) paragraph 6.2.3.II.D.1.f is extended to the passive failure of a pipe or a manual closed valve,
- All piping inside secondary containment was considered to remain intact to transport the leakage outside,
- Leakage paths having 2 normally closed valves and/or check valves in series were assumed to leak,
- Some leakage paths which required traveling a tortuous path were included, even though the path would allow plate-out and decay.
- No attempt was made to decouple the seismic event and LOCA, even though these coupled events have an extremely low probability.

Results

Bypass leakage was found to be credible only for the valves listed below in Table I. The detailed evaluation of individual leakage paths is included as Table II.

Table I

Component ID	Termination Area
AO-2(3)-203-2A MSIV	Turbine Building
AO-2(3)-203-2B MSIV	Turbine Building
AO-2(3)-203-2C MSIV	Turbine Building
AO-2(3)-203-2D MSIV	Turbine Building
2(3)-1301-17 & -20 Iso Condenser Vent Line	Turbine Building

Notes: (1) Leakage is released through the steam lines, to the turbine, and through the condenser.
(2) A new administrative limit is 46 scfh as the sum of the leak paths from Table I.

<u>Conclusion</u>

Secondary bypass leakage for both units has been evaluated and was shown to be limited to the MSIVs, and the Isolation Condenser vent line.

Secondary containment bypass leakage has been evaluated and is limited to the MSIVs and the Iso Condenser vent line isolation valves. Utilizing MSIV technical specification limit leakage for all four MSIVs and the Iso condenser vent line valves though the 30 day post accident period is conservative for bounding bypass leakage because containment post accident pressure falls to a few psig within 10 minutes. Therefore the actual leakage through the MSIVs will be significantly less than the technical specification limit over the 30 day post accident period.

TABLE II - DRESDEN 2 & 3 Potential Leakage Paths From Primary Containment Bypassing Secondary

Component ID	Termination Area	Comments
AO-2(3)-203-2A MSIV	TURBINE BUILDING	**Leak rate is accounted for**. Leakage through the MSIVs is directed to the condenser.
AO-2(3)-203-2B MSIV	TURBINE BUILDING	**Leak rate is accounted for.** Leakage through the MSIVs is directed to the condenser.
AO-2(3)-203-2C MSIV	TURBINE BUILDING	**Leak rate is accounted for.** Leakage through the MSIVs is directed to the condenser.
AO-2(3)-203-2D MSIV	TURBINE BUILDING	**Leak rate is accounted for.** Leakage through the MSIVs is directed to the condenser.
MO-2(3)-220-2 MS line drain	TURBINE BUILDING	**Leak rate is accounted for.** Leakage is not considered credible. Leakage from the drywell or torus has to pass through two normally closed valves. MO-2(3)-220-2 and MO-2(3)-220-1. Condensing steam will fill line forming a water seal.
2(3)-220-62A FW check valve	TURBINE BUILDING	Leakage is not considered credible. The Feedwater piping is water-filled.
2(3)-220-62B FW check valve	TURBINE BUILDING	Leakage is not considered credible. The Feedwater piping is water-filled.
AO-2(3)-1601-24 Drywell & Torus Vent	OUTSIDE	Leakage is not considered credible. Three normally closed fail closed valves are present or leakage would be captured by standby gas treatment.
3-8502-500 Nitrogen purge PCV Bypass	OUTSIDE	Leakage is not considered credible. The leakage has to pass through one fail-closed Group II Isolation valve and through a normally closed valve. Leakage would then travel through the nitrogen gas contained in the nitrogen vaporizer. Note: There is no Unit 2 counterpart.
AO-2(3)-1601-56 Torus inerting & purge	OUTSIDE	Leakage is not considered credible. There is a barrier of 3 closed valves or check valves.
AO-2(3)-1601-58 Torus inerting & purge	OUTSIDE	Leakage is not considered credible. There is a barrier of 3 closed valves or check valves.
AO-2(3)-1601-57 Drywell & Torus nitrogen makeup	OUTSIDE	Leakage is not considered credible. There is a barrier of 3 closed valves or check valves.
AO-2(3)-1601-55 Drywell & Torus inerting	OUTSIDE	Leakage is not considered credible. There is a barrier of 3 to 4 closed valves or check valves.
FCV-2(3)-8501-5B DW air sample vent return	HRSS	Leakage need not be considered for radiological consequences. HRSS is maintained at negative pressure, has its own charcoal filter unit, and is exhausted to the main stack.
FCV-2(3)-9205B DW air sample	HRSS	Leakage need not be considered for radiological consequences. HRSS is maintained at negative pressure, has its own charcoal filter unit, and is exhausted to the main stack.
FCV-2(3)-9206B DW air sample	HRSS	Leakage need not be considered for radiological consequences. HRSS is maintained at negative pressure, has its own charcoal filter unit, and is exhausted to the main stack.
FCV-2(3)-8501-3B DW air sample return	HRSS	Leakage need not be considered for radiological consequences. HRSS is maintained at negative pressure, has its own charcoal filter unit, and is exhausted to the main stack.
FCV-2(3)-8501-1B	HRSS	Leakage need not be considered for radiological consequences.

TABLE II - DRESDEN 2 & 3 Potential Leakage Paths From Primary Containment Bypassing Secondary

Component ID	Termination Area	Comments
Torus air sample		HRSS is maintained at negative pressure, has its own charcoal filter unit, and is exhausted to the main stack.
AO-2(3)-1601-60 Torus Vent	OUTSIDE	Leakage is not considered credible. Three normally closed fail closed valves are present or leakage would be captured by standby gas treatment..
AO-2(3)-1601-61 Torus Vent	OUTSIDE	Leakage is not considered credible. Three normally closed fail closed valves are present or leakage would be captured by standby gas treatment.
MO-2(3)-205-24 Reactor Head Cooling	TURBINE BUILDING	Leakage is not considered credible. Leakage is through head cooling piping through normally closed manual valve and 2 remote operated valves in series. This path is water-filled.
RVLIS Backfill System	TURBINE BUILDING	Leakage is not considered credible. Leakage is through Control Rod Drive piping which is water-filled. The RVLIS system is water filled.
AO-2-220-45 Reactor water sample	TURBINE BUILDING	Leakage is not considered credible. This path is water-filled and includes 3 normally closed valves. Note: There is no Unit 3 counterpart.
XCV-28941-721 (XCV-38941-721) HRSS sample	HRSS	Leakage need not be considered for radiological consequences. HRSS is maintained at negative pressure, has its own charcoal filter unit, and is exhausted to the main stack.
3/4" Lines 2(3)-0396A & B to CRD pumps - Seal water lines	TURBINE BUILDING	Leakage is not considered credible. Leakage is through Control Rod Drive piping to the drive pumps. This path is water filled.
MO-2(3)-1402-3A Core Spray pump suction	CST	Leakage is not considered credible. Leakage is through contaminated condensate storage fill line to the CST. This line is water-filled.
MO-2(3)-1402-3B Core Spray pump suction	CST	Leakage is not considered credible. Leakage is through contaminated condensate storage fill line to the CST. This line is water-filled.
Core Spray Discharge	RADWASTE, HRSS, TURBINE BUILDING	Leakage is not considered credible. Leakage must travel to drywell floor sump drain or equipment drain sump discharges. Multiple barriers exist between these sumps and the outside of secondary containment.
2-1301-505 (3-1302-500) Iso Condenser Steam Equalizing Line	TURBINE BUILDING	Leakage is not considered credible. Leakage from primary containment has to pass through two normally closed locked valves. Steam leakage would condense on the downstream side of the valve, causing a water trap to form.
MO-2(3)-1301-2 Iso Condenser steam supply	TURBINE BUILDING	Leakage is not a concern. This valve is normally open and must be open for a credible leak from valves AO-2(3)-1301-17, -20. (See next component below).
2(3)-1301-17 & -20 Iso Condenser Vent Line	TURBINE BUILDING	**Leak rate is accounted for.** The leakage is directed to the condenser.
MO-2(3)-1301-3 Iso Condenser condensate return	TURBINE BUILDING	Leakage is not considered credible. Leakage is through the Isolation Condenser return piping. Heat exchanger tube leak needs to occur to have a leakage path.
MO-2(3)-1501-	RADWASTE, HRSS,	Leakage is not considered credible. Leakage is to LPCI pump

TABLE II - DRESDEN 2 & 3 Potential Leakage Paths From Primary Containment Bypassing Secondary

Component ID	Termination Area	Comments
5A,B,C & D LPCI pump suction	CST	suction and through CST fill line to CST, RW, and HRSS. These leakage paths are water-filled.
RV-2-1599-13A,B,C & D (RV-3-1501-13A,B,C &D) LPCI suction relief	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is through relief lines to LPCI pump suction and through CST fill line to CST, RW, and HRSS. These leakage paths are water-filled.
MO-2(3)-1001-5A Shutdown Cooling return	HRSS	Leakage need not be considered for radiological consequences. HRSS is maintained at negative pressure, has its own charcoal filter unit, and is exhausted to the main stack.
MO-2(3)-1501-22A LPCI core flooding	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is along LPCI injection line through LPCI pumps to CST, RW, and HRSS. This path is water-filled.
MO-2(3)-1001-5B Shutdown Cooling return	HRSS	Leakage need not be considered for radiological consequences. HRSS is maintained at negative pressure, has its own charcoal filter unit, and is exhausted to the main stack.
MO-2(3)-1501-22B LPCI core flooding	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is along LPCI injection line through LPCI pumps to CST, RW, and HRSS. This path is water-filled.
MO-2(3)-1501-27B Containment spray	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is from LPCI spray header backwards through LPCI pumps to CST, RW, and HRSS. This path is water-filled.
MO-2(3)-1501-27A Containment spray	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is from LPCI spray header through LPCI pumps to CST, RW, and HRSS. This path is water-filled.
2(3)-1501-65A & B LPCI pump minimum flow	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is to LPCI pump discharge through the LPCI pumps to CST, RW, and HRSS. These paths are water-filled.
MO-2(3)-1501-38A LPCI test	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is along LPCI injection line through LPCI pumps to CST, RW, and HRSS. This path is water-filled.
AO-2(3)-1599-61, -62 Torus cross-tie to Hotwell	TURBINE BUILDING	Leakage is not considered credible. This path is water-filled.
2(3)-1501-65C & D LPCI pump minimum flow	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is to LPCI pump discharge and through the LPCI pumps to CST, RW, and HRSS. These paths are water-filled.
MO-2(3)-1501-38B LPCI test	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is along LPCI injection line through LPCI pumps to CST, RW, and HRSS. This path is water-filled.
MO-2(3)-2301-14 HPCI minimum flow	TURBINE BUILDING	Leakage is not considered credible. Leakage is along HPCI injection line through HPCI pumps to CST. This path is water-filled.
MO-2(3)-1501-18A LPCI Suppression Pool spray	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is along LPCI injection line through LPCI pumps to CST, RW, and HRSS. This path is water-filled.
MO-2(3)-1501-18B LPCI Suppression Pool spray	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is along LPCI injection line through LPCI pumps to CST, RW, and HRSS. This path is water-filled.
MO-2(3)-1201-2	TURBINE BUILDING	Leakage is not considered credible. Leakage is along RWCU

TABLE II - DRESDEN 2 & 3 Potential Leakage Paths From Primary Containment Bypassing Secondary

Component ID	Termination Area	Comments
RWCU supply		piping through a series of normally closed valves to RW and HRSS. This path is water-filled.
MO-2(3)-1201-3 RWCU supply	TURBINE BUILDING	Leakage is not considered credible. Leakage is along RWCU piping through a series of normally closed valves to RW and HRSS. This path is water-filled.
MO-2(3)-1001-2A & B Shutdown Cooling supply	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is through shutdown HX to LPCI system and then through valve MO-2(3)-1501-22A or -22B (see above). This path is water-filled.
MO-2(3)-1001-2C Shutdown Cooling supply	RADWASTE, HRSS, CST	Leakage is not considered credible. Leakage is through shutdown HX to LPCI system and then through valve MO-2(3)-1501-22A or -22B (see above). This path is water-filled.
2(3)-0399-506 CRD discharge to Rx recirc. loop	TURBINE BUILDING	Leakage is not considered credible. Leakage is along CRD discharge to Recirc. piping, through a series of normally closed manual isolation valves, to the Control Rod Drive water system. This path is water-filled.
2(3)-0301-95 CRD system	TURBINE BUILDING	Leakage is not considered credible. The CRD return line was removed from inside the drywell.
Various valves in the CRD Hydraulic Control Units	TURBINE BUILDING	Leakage is not considered credible. Leakage is through Control Rod Drive piping which is water-filled.
2(3)-4327-500 Demin. water supply	TURBINE BUILDING	Leakage is not considered credible. The Demineralized Water system piping is water -filled. Valves 2(3)-4327-500 and -502 are lock closed manual valves, and valves 2(3)-1916-500 and -501 are normally closed manual valves. Water is trapped in the piping bounded by these valves.
2(3)-4722 DW Instrument Air supply	OUTSIDE	Leakage is not considered credible. The Drywell Instrument Air piping is pressurized to 125 psig. Valve 2(3)-4722 is an air-operated diaphragm valve that is normally open and fails closed.
2(3)-4799-514 TIP Indexers Nitrogen Purge	TURBINE BUILDING	Leakage is not considered credible. Leakage would require the purge line isolation valves' automatic closure to fail and the purge check valve to fail. An instrument air line break in the Turbine Building would also be required.
2(3)-0733A, -B, -C, -D & -E TIP System Ball Valves	TURBINE BUILDING	Leakage is not considered credible. Leakage would require the purge line isolation valves' automatic closure to fail, the ball valves' automatic closure to fail and the manually operated cable shear valves to fail. An instrument air line break in the Turbine Building would also be required.
2-4799-521 DW Instrument Air to Instrument Air Cross-Tie	TURBINE BUILDING	Leakage is not considered credible. Leakage would have to pass through 1/4" line and normally closed 1/4" needle valve. Pump back system failure and an instrument air line break in the Turbine Building would also be required. Note: There is no Unit 3 counterpart.
2(3)-4721 DW Instrument Air suction	OUTSIDE	Leakage is not considered credible. Valves 2(3)-4720 and 2(3)-4721 are fail-closed air-operated diaphragm valves that are normally open but will be closed upon containment isolation signal. Inside the drywell, the system is a closed system.
2(3)-4640-500	TURBINE BUILDING	Leakage is not considered credible. Procedural controls (DTS

TABLE II - DRESDEN 2 & 3 Potential Leakage Paths From Primary Containment Bypassing Secondary

Component ID	Termination Area	Comments
DW Service Air Supply		1600-07) exist to verify that all the inboard and outboard service air valves are verified closed .
AO-2(3)-2001-106 DW Floor Drain sump discharge	RADWASTE, HRSS, TURBINE BUILDING	Leakage is not considered credible. The Drywell Floor Drain Sump discharge piping is water-filled. Valves AO-2(3)-2001-105 and -106 are air-operated diaphragm valves that are normally closed and fail-closed. Water is trapped in the piping bounded by these valves.
AO-2(3)-2001-6 DW Equipment Drain sump discharge	RADWASTE, HRSS, TURBINE BUILDING	Leakage is not considered credible. The Drywell Equipment Drain Sump discharge piping is water-filled. Valve 2(3)-2001-3 is an air operated gate valve that closes on Group II Isolation. Valves 2(3)-2001-5 and -6 are air operated diaphragm valves that are normally closed and fail-closed (on loss of instrument air). Water is trapped in the piping bounded by these valves.
MO-2(3)-2301-35 HPCI pump suction from torus	TURBINE BUILDING, CST	Leakage is not considered credible. Leakage is to HPCI pump suction and through CST fill line to CST. This path is water-filled.
MO-2(3)-2301-5 HPCI steam supply	TURBINE BUILDING	Leakage is not considered credible. Leakage Path is through the Condensate Drain Lines. Valves 2(3)-2301-29 and 30 are air operated globe valves that fail closed. Condensate will collect in the drain line providing a water seal.
2(3)-2301-34 HPCI condensate drain	TURBINE BUILDING	Leakage is not considered credible. The HPCI Condensate Drain piping is normally water-filled below the normal torus water level.
2(3)-2599-23B ACAD dilution air supply to DW	U3 RB	Leakage is not considered credible. Leakage is back through compressor cross-tie to Unit 3 drywell (assuming U3 (U2) outage) and ends in RB.
2(3)-2599-24B ACAD dilution air supply to DW	U3 RB	Leakage is not considered credible. Leakage is back through compressor cross-tie to Unit 3 drywell (assuming U3 (U2) outage) and ends in RB.

Appendix C - Quad Cities Bypass Leakage Evaluation

Purpose

The purpose of this evaluation is to validate that the 46 scfh (at 25 psig) main steam isolation valve (MSIV) leakage is the bounding leakage value for all leakage paths for purposes of dose to the control room operators after a loss of coolant accident (LOCA). The criteria and methodology are listed below.

Background

Pathways from primary containment to points outside of secondary containment boundary which bypass secondary containment (and the Standby Gas Treatment System) exist at Quad Cities Station. With the exception of the MSIV leakage pathway, these potential bypass pathways have not been explicitly considered (assigned a separate leakage value) in the radiological assessments of LOCA scenarios. SRP guidance for evaluating Control Room dose requires a review of all potential bypass pathways.

Criteria

All penetrations of primary containment were included in the evaluation. The following criteria and assumptions were used to evaluate if the lines need to be considered for secondary containment bypass leakage:

1. Potential leakage paths that terminate in the secondary containment are excluded.
2. All valves were assumed to leak, however, 3 or more closed valves in series would result in negligible leakage, allowing the line to be excluded.
3. Water-filled paths can be eliminated based on additional scrubbing in the water.
4. Lack of credible driving force 10 minutes after a LOCA (concern is for first 10 min.). Ref. UFSAR Fig. 6.3 -42.
5. The Standby Gas Treatment System is operating.
6. The Emergency Core Cooling System pumps are operating.
7. The HRSS Ventilating System remains intact following a LOCA.

Basis

The basis for the evaluation methodology is from guidance provided by the NRC and is as follows:

1. The Standard Review Plan (SRP) 15.6.5, Appendix A states (middle of page 15.6.5-10) that the secondary containment bypass leakage rate must be considered. Also (page 15.6.5-10, bottom): "For dual containment systems, the bypass leakage is evaluated. This leakage, usually expressed as a fraction or percentage of the primary containment leak rate, is assumed to pass from primary containment directly to the environment, bypassing the secondary containment."

2. SRP 6.2.3 (pg. 6.2.3-5) states "The fraction of primary containment leakage bypassing secondary containment and escaping directly to the environment should be specified. Branch Technical Position (BTP) CSB 6-3 provides guidance ..."

3. BTP CSB 6-3 (pg. 6.2.3-11, item 6) states "The total leakage rate for all potential bypass leakage paths ... **should be determined in a realistic manner** ... This value should be used in calculating the off-site radiological consequences of postulated loss of coolant accidents and in setting technical specification limits with margin for bypass leakage."

The NRC has not provided any specific guidance to quantify the "realistic manner" in which the bypass leakage is to be determined. A review of two cases is as follows: In 1976, a question was asked of license applicants to identify and quantify potential bypass leakage, LaSalle responded with a line list and arguments that multiple isolation valves, water filled legs of piping, et cetera were sufficient to preclude any bypass leakage in its design. This was accepted by the NRC. Fermi-2 responded similarly except that they stated conservatively, 4% of the containment leakage (0.5%/day) would bypass secondary containment; the NRC accepted this and included it in their SER assessment.

The HRSS Ventilation System is assumed to remain intact for several reasons. First, non-seismic ducting/piping does not fail in a seismic event due to its inherent ruggedness. The ducting/piping may deform, but no loss of pressure boundary occurs. Second, Loss of Offsite Power (LOOP) will reduce airflow through the HRSS Ventilation System. The HRSS Ventilation System fans lose power during a LOOP, but air will still flow through the system since air will still be flowing out the main plant vent stack - the 'chimney effect' will draw the air through. Third, a tornado that could completely disable the HRSS Ventilation System is not postulated to occur simultaneously with a LOCA.

Methodology

The evaluation considered all system piping routes connected to primary containment that also penetrate the secondary containment boundary. Further details on credibility of individual leakage path potential are shown in the comments of Table II.

The methodology includes the following sources of conservatism:

- In many cases, the criterion of single active failure stated in NUREG-0800 (7/81) paragraph 6.2.3.II.D.1.f is extended to the passive failure of a pipe or a manual closed valve,
- All piping inside secondary containment was considered to remain intact to transport the leakage outside,
- Leakage paths having 2 normally closed valves and/or check valves in series were assumed to leak,
- Some leakage paths required traveling a tortuous path, which would allow plate-out and decay.
- No attempt was made to decouple the seismic event and LOCA. Even though these coupled events have an extremely low probability.

Results

Bypass leakage was found to be credible only for the valves listed below in Table I. The detailed evaluation of individual leakage paths is included as Table II.

Table I

Component ID	Termination Area
AO-2-203-2A MSIV	Turbine Building
AO-2-203-2B MSIV	Turbine Building
AO-2-203-2C MSIV	Turbine Building
AO-2-203-2D MSIV	Turbine Building

Notes: Leakage is released through the steam lines, through the turbine, and through the condenser.

Conclusion

Secondary Containment bypass leakage has been evaluated and is limited to the MSIVs. Utilizing MSIV Technical Specification limit leakage for all four MSIVs through the 30 day post accident period is conservative for bounding bypass leakage because containment post accident pressure falls to a few psig within 10 minutes. Therefore the actual leakage through the MSIV's will be significantly less than the technical specification limit over the 30 day post accident period.

TABLE II - QUAD CITIES 1&2 - Potential Leakage Paths From Primary Containment Bypassing Secondary Containment

Component ID	Termination Area	Comments
AO-1(2)-203-2A MSIV	Turbine Building	**Leak rate is accounted for.** Leakage through the MSIVs is directed to the condenser.
AO-1(2)-203-2B MSIV	Turbine Building	**Leak rate is accounted for.** Leakage through the MSIVs is directed to the condenser.
AO-1(2)-203-2C MSIV	Turbine Building	**Leak rate is accounted for.** Leakage through the MSIVs is directed to the condenser.
AO-1(2)-203-2D MSIV	Turbine Building	**Leak rate is accounted for.** Leakage through the MSIVs is directed to the condenser.
MO-1(2)-220-2 Main Steam line drain	Turbine Building	Leakage is not considered credible. Leakage through the MSL Drain valves is directed to the condenser. Condensing steam will fill line forming a water seal.
1(2)-220-62A FW check valve	Turbine Building	Leakage is not considered credible. The line is water filled.
1(2)-220-62B FW check valve	Turbine Building	Leakage is not considered credible. The line is water filled.
DW Pneumatic, Inst. Air & N2 Backup (Suction)	Turbine Building, Outside	Leakage is not considered credible. Instrument Air and Nitrogen Systems are backup supplies to the Drywell Pneumatic System. Leakage path would be through AO-1(2)-4720 and AO-1(2)-4721 which both fail closed. It would then have to travel through a compressor, separator, filter, dryer, 250 gal. receiver tank and one of two parallel paths. Normally closed manual valve 1(2)-4799-168 or backwards through check valve 1(2)-4799-166 and PCV 1(2)-4723 which fails closed. DW pressure would have to be greater than Instrument Air or N2 pressure.
DW Pneumatic, Inst. Air & N2 Backup (Supply)	Turbine Building, Outside	Leakage is not considered credible. Instrument Air and Nitrogen Systems are backup supplies to the Drywell Pneumatic System. Parallel leakage paths exist backwards through check valves 1(2)-4799-155 and 1(2)-4799-156 or backwards through check valves 1(2)-4799-158 and 1(2)-4799-159. It would then have to travel through AO-1(2)-4722A or AO-1(2)-4722B and one of two parallel paths. Normally closed manual valve 1(2)-4799-168 or backwards through check valve 1(2)-4799-166 and PCV 1(2)-4723 which fails closed. DW pressure would have to be greater than Instrument Air or N2 pressure.
1(2)-4699-46 Service Air to DW	Turbine Building	Leakage is not considered credible. Flow would be backwards through a check valve and closed manual valve. Procedural controls, QCOP 1600-11, exist to verify this valve is closed.
AO-1(2)-1601-23 AO-1(2)-1601-62 AO-1(2)-1601-60 AO-1(2)-1601-61 Drywell & Torus Vent	Outside	Leakage is not considered credible. Leakage path is through at least 3 normally closed, fail-closed valves in series or to SBGT.
AO-1(2)-2599-4A ACAD	SBGT	Leakage is not considered credible. These valves are normally closed and have no auto-open functions. They are used to vent the Drywell to the operating SBGT System after an accident.
AO-1(2)-2599-4B ACAD	SBGT	Leakage is not considered credible. These valves are normally closed and have no auto-open functions. They are used to vent the Drywell to the operating SBGT System after an accident.
AO-1(2)-1601-21	1/2 EDG Room and Outside	Leakage is not considered credible. Through AO-1(2)-1601-55 to Nitrogen System and Pump-back Air System (Joy A/C). The Nitrogen System is located in the 1/2 EDG room and outside. Leakage has to pass through one fail-closed Group II valve, AO-1(2)-1601-21 or 56, and then through valve AO-1(2)-1601-55 which also closes on a Group II signal. It also would have to pass through normally closed manual valves 1(2)-8799-16 or 17. The Pump-back Air System does not leave the Reactor Building. AO-1(2)-1601-22 and -56 open to the Reactor Building.

TABLE II - QUAD CITIES 1&2 - Potential Leakage Paths From Primary Containment Bypassing Secondary Containment

Component ID	Termination Area	Comments
AO-1(2)-1601-59	1/2 EDG Room and Outside	Leakage is not considered credible. Through AO-1(2)-1601-57 to Nitrogen System and Pump-back Air System (Joy A/C). The Nitrogen System is located in the 1/2 EDG room and outside. Leakage has to pass through one fail-closed Group II valve, AO-1(2)-1601-58 or 59, and then through valve AO-1(2)-1601-57 which also closes on a Group II signal. It also has to pass through one of two parallel paths, PCV-1(2)-8799-20 which is in manual and closed or auto with the setpoint >1.0 psi. Or through closed manual valve 1(2)-8799-21. When drywell pressure is above the setpoint, this would provide three closed valves in series. The Pump-back Air System does not leave the Reactor Building.
1(2)-1601-76	Rx Building	Leakage is not considered credible. Return from Primary Containment Oxygen Sampling System, does not leave the Reactor Building.
AO-1(2)-1601-58	1/2 EDG Room and Outside	Leakage is not considered credible. AO-1(2)-1601-58 through AO-1(2)-1601-57 to Nitrogen System and Pump-back Air System (Joy A/C). See AO-1(2)-1601-59 explanation above.
AO-1(2)-1601-20A	Rx Building	Leakage is not considered credible. AO-1(2)-1601-20A open to the Reactor Building.
AO-1(2)-1601-20B	Rx Building	Leakage is not considered credible. AO-1(2)-1601-20B open to the Reactor Building.
AO-1(2)-8801D Torus Air Sample	HRSS	Leakage is not considered credible. See component ID AO-1-8801 A/D.
1(2)-0263-943A 1(2)-0263-946A RVLIS Backfill	Turbine Building	Leakage is not considered credible. Leakage is through two check valves and at least one needle valves all in series in the Control Rod Drive piping which is water-filled.
1(2)-0263-943B 1(2)-0263-946B RVLIS Backfill	Turbine Building	Leakage is not considered credible. Leakage is through two check valves and at least one needle valves all in series in the Control Rod Drive piping which is water-filled.
MO-1(2)-1201-5 RWCU Supply	Turbine Building & Radwaste Building	Leakage is not considered credible. See MO-1(2)-1201-5 RWCU Supply.
AO-1(2)-220-45 Reactor Water Sample	HRSS Building & Turbine Building	Leakage is not considered credible. The system is water filled.
MO-1(2)-1402-25A Core Spray Injection	Turbine Building	Leakage is not considered credible. The system is water filled.
MO-1(2)-1402-25B Core Spray Injection	Turbine Building	Leakage is not considered credible. The system is water filled.
MO-1(2)-1402-3A Core Spray Torus Suction	Turbine Building & Outside	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1402-3B Core Spray Torus Suction	Turbine Building & Outside	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-47 Shutdown Cooling	Turbine Building, Radwaste	Leakage is not considered credible. All paths are normally closed and water filled.
MO-1(2)-1001-7A/B RHR Torus Suction	Radwaste, HRSS, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-7C/D RHR Torus Suction	Radwaste, HRSS, CCST	Leakage is not considered credible. This path is water-filled.
RV 1(2)-1001-125A/D RHR Suction Relief	Radwaste, HRSS, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-20 &21	Turbine Building and	Leakage is not considered credible. This path is water-filled.

TABLE II - QUAD CITIES 1&2 - Potential Leakage Paths From Primary Containment Bypassing Secondary Containment

Component ID	Termination Area	Comments
RHR X-tie to Radwaste	Radwaste Building	
1(2)-1001-131 &132 Keep Fill System	Turbine Building, CCST	Leakage is not considered credible. This path is water filled.
MO-1(2)-2301-36 HPCI Torus Suction	Turbine Building, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1301-25 RCIC Torus Suction	Turbine Building, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-36A Torus Cooling	Turbine Building, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-18A RHR Min Flow	Turbine Building, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1402-4A/B Core Spray Test MO-1(2)-1402-38A/B Core Spray Min Flow	Turbine Building, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-2301-14 HPCI Min Flow	Turbine Building, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1301-60 RCIC Min Flow	Turbine Building, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-36B Torus Cooling	Turbine Building, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-18B RHR Min Flow	Turbine Building, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-37A Torus Spray	Radwaste, HRSS, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-37B Torus Spray	Radwaste, HRSS, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-29A LPCI Injection	Radwaste, HRSS, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-29B LPCI Injection	Radwaste, HRSS, CCST	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1001-26A Containment Spray	Outside	Leakage is not considered credible. Line 1(2)-1029A-10" is blind flanged in the Reactor Building and only used during IPCLRT which is performed with the unit shutdown. Leakage path is water-filled.
MO-1(2)-1001-26B Containment Spray	Outside	Leakage is not considered credible. This leakage path is water-filled.
Fuel Pool Cooling Assist Return	Radwaste	Leakage is not considered credible. Pipe spool piece removed during plant operations.
Fuel Pool Cooling Assist Suction	Radwaste	Leakage is not considered credible. Pipe spool piece removed during plant operations.
RHR Drains to Radwaste	Turbine Building, Radwaste	Leakage is not considered credible. Lines are water filled and locked close.
Various valves in the CRD Hydraulic Control Units	Turbine Building	Leakage is not considered credible. Leakage is through Control Rod Drive piping which is water-filled.
MO-1/2-7504A/B SBGT	Turbine Building	Leakage is not considered credible. Leakage path is from Drywell or torus vent through MO-1/2-7504A/B to the Turbine Building. During Drywell or torus venting the SBGT System would be operating creating a negative pressure on the Reactor Building side of this valve therefore, leakage would be from the Turbine Building into the Reactor Building or SBGT System.
MO-1(2)-2301-5	Turbine	Leakage is not considered credible. Leakage path is from the HPCI steam supply

TABLE II - QUAD CITIES 1&2 - Potential Leakage Paths From Primary Containment Bypassing Secondary Containment

Component ID	Termination Area	Comments
HPCI Steam Supply	Building	condensate drain to the main condenser. AO-1(2)-2301-29 and AO-1(2)-2301-30 close and AO-1(2)-2301-28 opens on HPCI initiation. This isolates the path to the condenser and allows the condensate to drain to the torus.
1(2)-2301-34 HPCI Condensate Drain	Turbine Building	Leakage is not considered credible. The HPCI condensate drain piping is normally water filled below the normal torus water level.
MO-1(2)-1201-5 RWCU Supply	Turbine Building, Radwaste	Leakage is not considered credible. This path is water-filled.
MO-1(2)-1301-17 RCIC Steam Supply	Turbine Building	Leakage is not considered credible. Leakage path is from the RCIC steam supply condensate drain to the main condenser. AO-1(2)-1301-34 and AO-1(2)-1301-35 close on RCIC initiation. This isolates the path to the condenser and the drain line will collect with condensate.
2-4399-45 Clean Demin to DW	Turbine Building	Leakage is not considered credible. Manual valve to drywell is closed during normal operations. Procedural controls, QCOP 1600-11, exist to verify this valve is closed.
AO-1(2)-8801A/D DW & Torus Air Sample	HRSS	Leakage is not considered credible. Leakage path would be through three normally closed valves. Leakage need not be considered for radiological consequences since HRSS is maintained at negative pressure, has its own charcoal filter unit, and is exhausted to the main stack.
SO 1(2)-0799-3D TIP Indexer Purge Air Supply	Turbine Building	Leakage is not considered credible. Leakage from the 3/8" line would require the purge line isolation valve, SO 1(2)-799-3D, automatic closure to fail and the purge check valve, 1(2)-743, to fail. An instrument air line break in the Turbine Building or loss of instrument air compressors would also be required.
AO-1(2)-2001-3 Drywell Floor Drain Sump	Radwaste, HRSS	Leakage is not considered credible. This path is water-filled.
AO-1(2)-2001-15 Drywell Equipment Drain Sump	Radwaste, HRSS	Leakage is not considered credible. This path is water-filled.

Appendix - D Figures

Figure 1

Dresden Revised Radiological Control Room Model

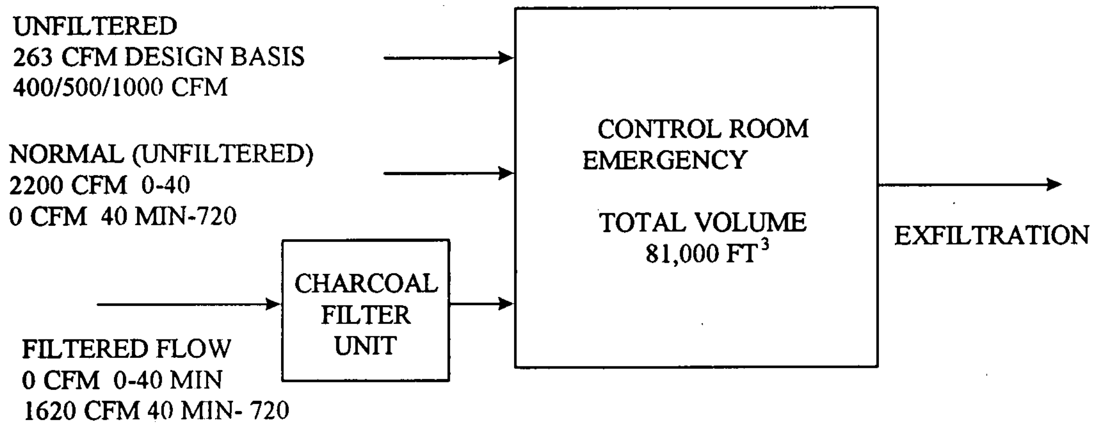


Figure 2

Quad Cities Revised Radiological Control Room Model

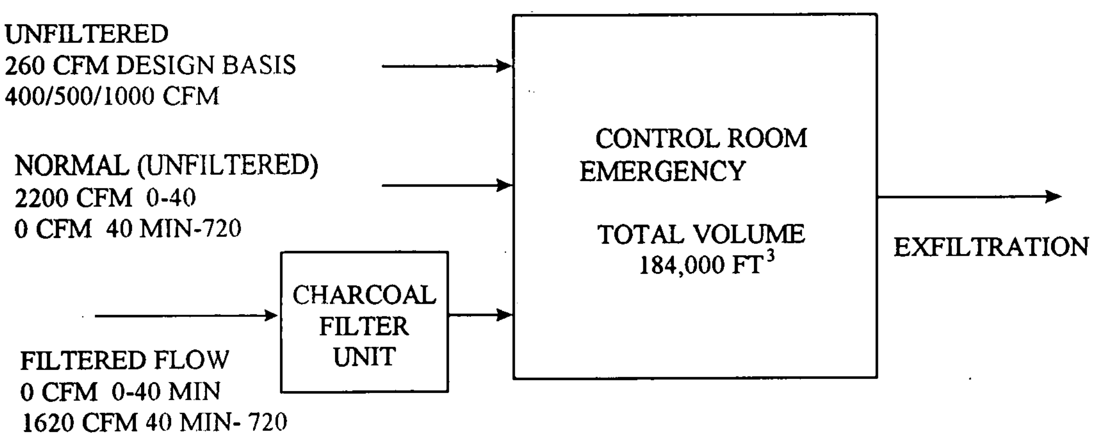


FIGURE 3
DRESDEN CR OPERATOR THYROID DOSE VS INFILTRATION
(46 SCFH MSIV LEAKAGE)

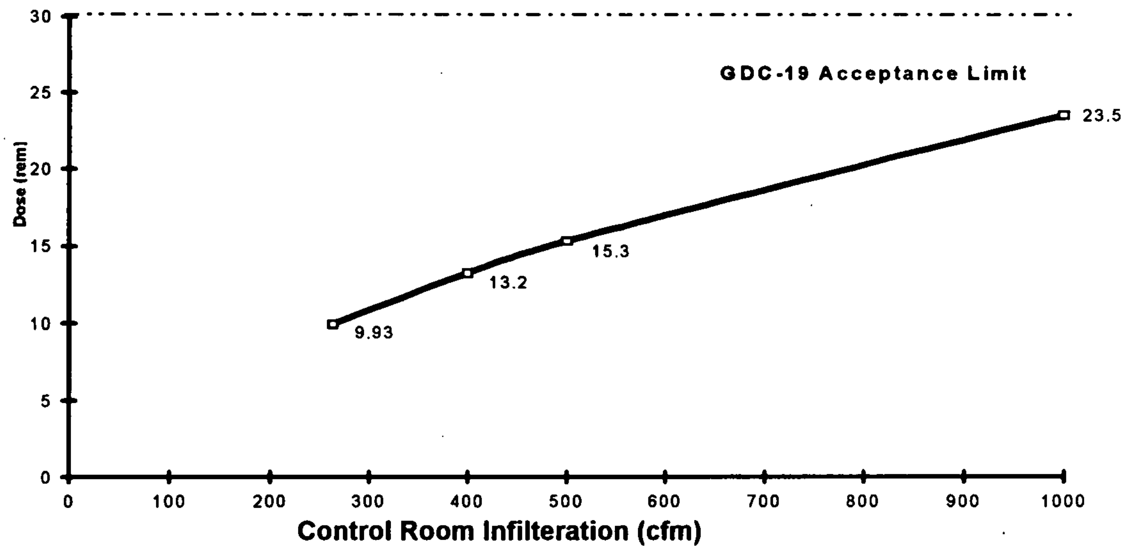


FIGURE 4
DRESDEN CR OPERATOR THYROID DOSE VS INFILTRATION
(120 SCFH MSIV LEAKAGE)

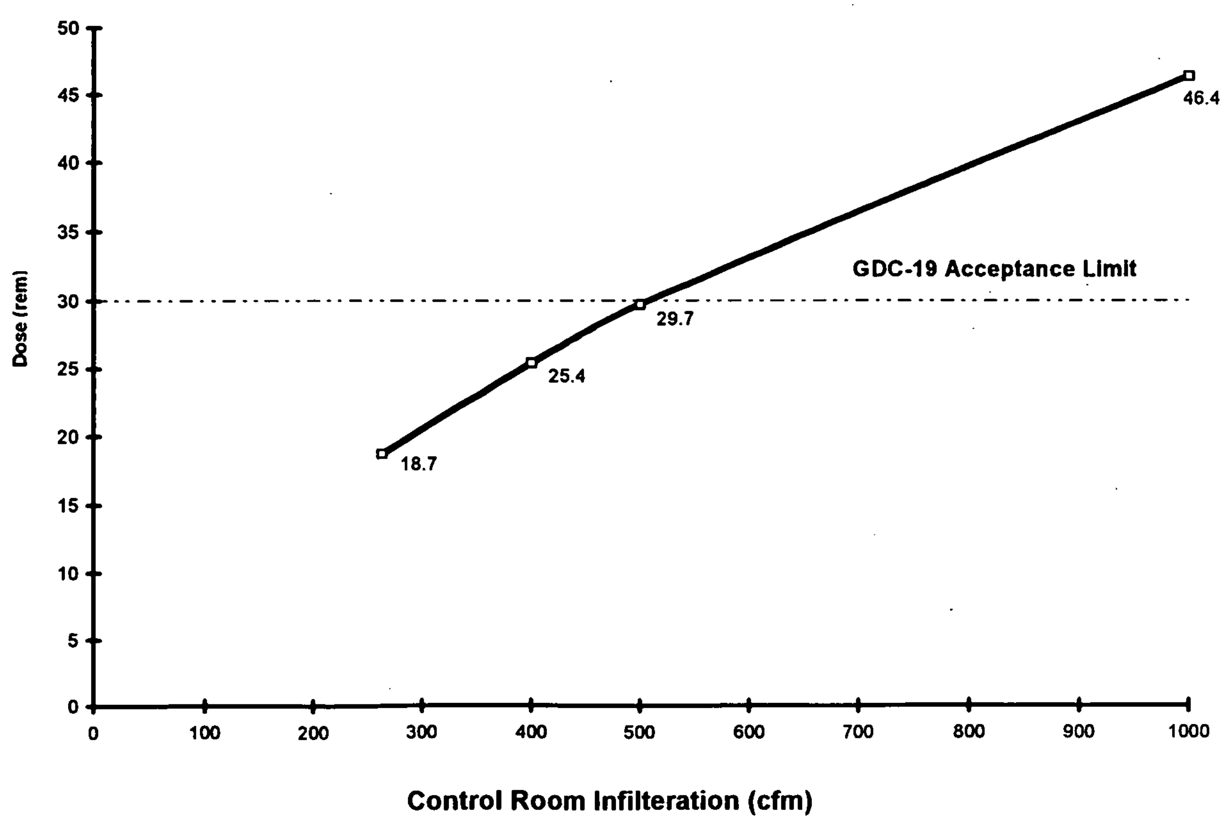


FIGURE 5
QUAD CITIES CR OPERATOR THYROID DOSE VS INFILTRATION
(46 SCFH MSIV LEAKAGE)

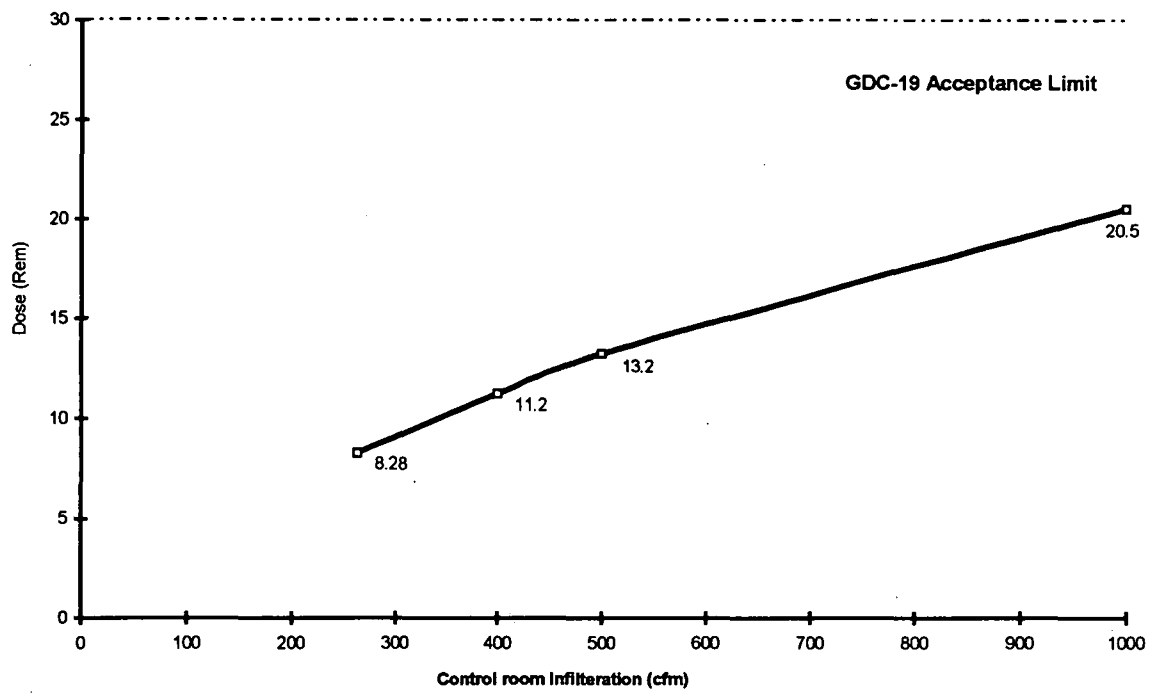
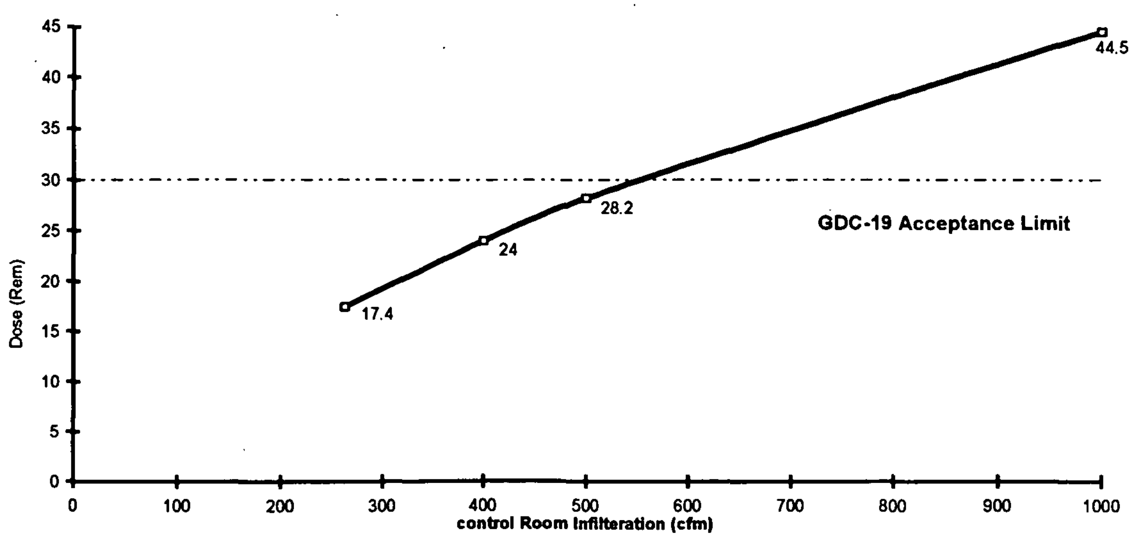


FIGURE 6
QUAD CITIES CR OPERATOR THYROID DOSE VS INFILTRATION
(120 SCFH MSIV LEAKAGE)



Appendix - E Tables

Table 1

Assumptions and Inputs Used in Control Room Habitability Radiological Analyses

Assumption/Input	Current Dresden Value	Revised Dresden Value	Current Quad Cities Value	Revised Quad Cities Value	Comments on Revised Value
I. Data and Assumptions Used To Estimate Radioactive Source from a Postulated Loss-of-Coolant Accident					
A. Power Level, MWt	2527	2578	2511	2561	Power level increased to 102% for this reanalysis to be consistent with SRP 15.6.5.
B. Operating History and Source Term	1000 days TID 14844*	1000 days using TID 14844 (909 2727 days using Siemens Fuel)**	1000 days TID 14844*	1000 days using TID 14844 (909 to 2727 days using Siemens Fuel)**	*SRP 15.6.5 Recommendation ** Also assessed the impact on the CR dose using the New Siemens fuel (with a 2-year cycle). For the 24 mo. cycle Siemens fuel the analysis considerded a Burnup of 60,000 Mwd/MTU
C. Fission Products Released from Damaged Fuel Noble Gases Halogens	 100% 25%	 100% 25%	 100% 25%	 100% 25%	Reg. Guide 1.3 and SRP 15.6.5
D. Iodine Fractions Organic Elemental Particulate	 0.04 0.91 0.05	 0.04 0.91 0.05	 0.04 0.91 0.05	 0.04 0.91 0.05	Reg. Guide 1.3 and SRP 15.6.5
E. Fission Product Scrubbing	Not Used	DF=5	Not Used	DF=5	Fission product scrubbing in the suppression pool per SRP 6.5.5 (Mk I containment)
F. ESF Leakage Outside Primary Containment	Not Used	10 gph (2 x 5 gph)	Not Used	10 gph (2 x 5 gph)	ESF leakage is included per SRP15.6.5, Appendix B
G Fission Product Released to Suppression Pool	Not Used	50%	Not Used	50%	SRP 15.6.5
H. Suppression Pool Volume	Not Used	110,000 ft^3	Not Used	110,000 ft^3	Conservative bounding value

TABLE 3
Control Room LOCA Dose Calculation Results
Quad Cities Units 1 & 2

The original design basis was rerun to ensure proper AXIDENT code function and inputs. The results agree with NUS Calc. 546Y-M-10 dose with only a 0.13 rem difference. This minor difference is due to input differences. The previous calculation was run with a control room isolation time of 120 minutes (2 hrs) while this calculation used 110 minutes. The results of this run are as follows:

Original Design Basis

Leakage Path	Thyroid Dose (rem)	
SBGTS Contribution		
Unfiltered Mode	15.50	
Filtered Mode	7.63	
MSIV Contribution		
Unfiltered Mode	1.15E-1	
Filtered Mode	6.03	
Total	29.28	= 29.3 rem

The CR dose from the revised design basis analysis using the updated methodologies are as follows:

Revised Design Basis

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	8.13E-1
Filtered Mode	1.57
MSIV Contribution	
Unfiltered Mode	1.07E-2
Filtered Mode	5.81
ESF Contribution	
Unfiltered Mode	2.47E-2
Filtered Mode	4.84E-2
Total	8.28

TABLE 3
Control Room LOCA Dose Calculation Results
Quad Cities Units 1 & 2

The whole body and beta doses for the original design basis case are identical to those previously calculated in the original dose analysis, NUS Calculation 546Y-M-07/S1, and NUS Calculation 546Y-M-10 with the exception of the MSIV contribution. A typographical error in the X/Q input for the last time step (8.82E-5 vs. 8.52E-5) in Calc. 546Y-M-07/S1 produced 1.18E-2 vs 1.15E-2 for the whole body dose, however the total whole body dose remained at 1.18E-1 rem. A beta dose of 0.476 vs. 0.465 was produced, therefore the actual beta dose is 1.22 not 1.23 rem. While the whole body and beta doses for the revised design basis are slightly higher than the original design basis values, they are well within the 5 rem and 30 rem regulatory limits.

Original Design Basis

	Whole Body (rem)	Beta (rem)
SBGTS Contribution		
CR Activity	3.04E-2	7.53E-1
Plume Shine	1.66E-2	
MSIV Contribution		
CR Activity	1.15E-2	4.65E-1
Plume Shine	2.03E-3	
Facility Shine	5.70E-2	
Total	1.18E-1	1.22

Revised Design Basis

	Whole Body (rem)	Beta (rem)
SBGTS Contribution		
CR Activity	7.35E-2	1.68
Plume Shine	1.66E-2	
MSIV Contribution		
CR Activity	1.33E-2	5.36E-1
Plume Shine	2.03E-3	
ESF Contribution	5.18E-4	1.18E-2
Facility Shine	5.70E-2	
Total	1.63E-1	2.23

TABLE 3
Control Room LOCA Dose Calculation Results
Quad Cities Units 1 & 2

The control room operator thyroid dose is reduced from 29.3 rem to 8.28 rem using the revised methodologies with 260 scfm infiltration. The calculated control room dose for the cases with higher control room infiltration rates are as follows:

Revised Design Basis w/400 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	8.01E-1
Filtered Mode	2.20
MSIV Contribution	
Unfiltered Mode	1.06E-2
Filtered Mode	8.14
ESF Contribution	
Unfiltered Mode	2.43E-2
Filtered Mode	6.79E-2
Total	11.2

Revised Design Basis w/500 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	7.94E-1
Filtered Mode	2.60
MSIV Contribution	
Unfiltered Mode	1.05E-2
Filtered Mode	9.64
ESF Contribution	
Unfiltered Mode	2.41E-2
Filtered Mode	8.04E-2
Total	13.2

Revised Design Basis w/1000 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	7.67E-1
Filtered Mode	4.16
MSIV Contribution	
Unfiltered Mode	1.01E-2
Filtered Mode	15.4
ESF Contribution	
Unfiltered Mode	2.33E-2
Filtered Mode	1.28E-1
Total	20.5

TABLE 3
Control Room LOCA Dose Calculation Results
Quad Cities Units 1 & 2

With an alternate MSIV leak rate of 120 scfh instead of 46 scfh, the following calculated control room dose for the full range of control room infiltration rates (260, 400, 500, and 1000 cfm) are higher.

Revised Design Basis w/120 scfh MSIV and 260 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	8.13E-1
Filtered Mode	1.57
MSIV Contribution	
Unfiltered Mode	2.81E-2
Filtered Mode	14.9
ESF Contribution	
Unfiltered Mode	2.47E-2
Filtered Mode	4.84E-2
Total	17.4

Revised Design Basis w/120 scfh MSIV and 400 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	8.01E-1
Filtered Mode	2.20
MSIV Contribution	
Unfiltered Mode	2.76E-2
Filtered Mode	20.9
ESF Contribution	
Unfiltered Mode	2.43E-2
Filtered Mode	6.79E-2
Total	24.0

Revised Design Basis w/120 scfh MSIV and 500 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	7.94E-1
Filtered Mode	2.60
MSIV Contribution	
Unfiltered Mode	2.74E-2
Filtered Mode	24.7
ESF Contribution	
Unfiltered Mode	2.41E-2
Filtered Mode	8.04E-2
Total	28.2

TABLE 3
Control Room LOCA Dose Calculation Results
Quad Cities Units 1 & 2

Revised Design Basis w/120 scfh MSIV and 1000 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	7.67E-1
Filtered Mode	4.16
MSIV Contribution	
Unfiltered Mode	2.64E-2
Filtered Mode	39.4
ESF Contribution	
Unfiltered Mode	2.33E-2
Filtered Mode	1.28E-1
Total	44.5

TABLE 3A
Control Room LOCA Dose Calculation
Siemens Fuel
Quad Cities Units 1 & 2

The calculated control room dose from the Siemens fuel with a 20,000 Mwd/MTU burnup are approximately 6% higher than with the TID fuel. The maximum depletion with 60,000 Mwd/MTU burnup yields a 15.5% higher dose.

Revised Design Basis w/20,000 Mwd/MTU Siemens Fuel

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	8.39E-1
Filtered Mode	1.64
MSIV Contribution	
Unfiltered Mode	1.11E-2
Filtered Mode	6.21
ESF Contribution	
Unfiltered Mode	2.54E-2
Filtered Mode	5.06E-2
Total	8.78

Revised Design Basis w/60,000 Mwd/MTU Siemens Fuel

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	8.96E-1
Filtered Mode	1.77
MSIV Contribution	
Unfiltered Mode	1.18E-2
Filtered Mode	6.80
ESF Contribution	
Unfiltered Mode	2.72E-2
Filtered Mode	5.47E-2
Total	9.56

TABLE 5
Offsite LOCA Dose Calculation Results
Dresden Units 2 & 3 and Quad Cities Units 1 & 2

DRESDEN

EAB Dose Comparison

	Thyroid (rem)	Wholebody (rem)
Original SER	185	8
Revised methodology (no mixing)	6.36	5.91
Revised Methodology (mixing)	0.47	0.30

LPZ Dose Contribution

	Thyroid (rem)	Wholebody (rem)
Original SER	90	2
Revised methodology (no mixing)	5.64	1.39
Revised methodology (mixing)	3.64	0.32

QUAD CITIES

EAB Dose Comparison

	Thyroid (rem)	Wholebody (rem)
Original SER	150	6
Revised methodology (no mixing)	6.68	6.7
Revised methodology (mixing)	0.46	0.26

LPZ Dose Contribution

	Thyroid (rem)	Wholebody (rem)
Original SER	108	3
Revised methodology (no mixing)	5.84	1.46
Revised methodology (mixing)	3.71	0.33

Table 1

Assumptions and Inputs Used in Control Room Habitability Radiological Analyses

• Assumption/Input		• Current Dresden Value	• Revised Dresden Value	• Current Quad Cities Value	• Revised Quad Cities Value	• Comments on Revised Value
II.	**Data and Assumptions Used To Estimate Activity Released From Loss-of-Coolant Accident**					
A.	Primary Containment Leak Rate - total, %/day	1.6	1.6	1.0	1.0	Per Reg. Guide 1.3 and SRP 15.6.5, value in Technical Specifications
B.	Leak Rate through each MSIV, scfh @ 25 psig @ 48 psig*	 11.5 18.2	 11.5 19.9	 11.5 18.2	 11.5 19.9	Per Reg. Guide 1.3 and SRP 15.6.5 *Change in extrapolation factor from 1.58 to 1.73 using ORNL NSIC-5
C.	Total Leak Rate: 4 steam lines, scfh @ 25 psig @ 48 psig*	 46.0 72.7	 46.0/120.0 79.6/207.6	 46.0 72.7	 46.0/120.0 79.6/207.6	Higher total rates analyzed for possible future tech spec change *ORNL NSIG-5 extrapolation factor = 1.73
D.	Volume of Primary Containment (mixed volume), cu. ft.	275,000	278,000*	275,000	269,000**	*GE Document Net-2300-740, ** QC UFSAR Table 6-2-1
E.	Primary Containment Leak Rate which Goes to Secondary, %/Day	1.45	1.44	0.85	0.83	Calculated – ILRT tech. spec. less MSIV leakage varying from 11.5 scfh per valve to 30 scfh per valve
F.	Primary Containment Leak Rate which goes through MSIV, %/Day	0.15*	0.16 up to 0.42**	0.15*	0.17 up to 0.44**	*Calculated value **Exploratory analysis varying from 11.5 scfh per value to 30 scfh per value
G.	SBGTS Adsorption(1) and Filtration Efficiencies, % Organic Iodines Elemental Iodines Particulate Iodines	 90 90 90	 95% 95% 95%	 90 90 90	 95% 95% 95%	Per new Tech Specs for Quad Cities and Tech Spec change request for Dresden
H.	Secondary Containment Release Leak Rate, %/Day	100	310%	100	297%	Considers recirc mixing in 50% of volume, 10% tech spec tolerance and 10% margin (new calculated volume used)
I.	Leak Rate from Turbine-Condenser Complex, %/Day	1	1	1	1	Original III.D.3.4 submittal

Table 1
Assumptions and Inputs Used in Control Room Habitability Radiological Analyses

	• Assumption/Input	• Current Dresden Value	• Revised Dresden Value	• Current Quad Cities Value	• Revised Quad Cities Value	• Comments on Revised Value
J.	Plateout Removal Constant - MSIV Leak Rate Only, Sec-1					Calc. Based on Description in Original III.D.3.4 submittal
	Elemental Iodine	1.503 E-3	1.503 E-3	1.503 E-3	1.503 E-3	
	Particulate Iodine	1.503 E-3	1.503 E-3	1.503 E-3	1.503 E-3	
	Organic Iodine	0	0	0	0	
K.	Dispersion Data - at intake, sec/m^3					From Original III.D.3.4 submittal - utilizing the Halitsky methodology for the ground level release and Regulatory Guide 1.3 for the SGBTS Release
	MSIV Leakage					
	0-2 hr	1.29 E-3	1.29 E-3	1.29 E-3	1.29 E-3	
	2-8 hr	1.29 E-3	1.29 E-3	1.29 E-3	1.29 E-3	
	8-24 hr	7.61 E-4	7.61 E-4	7.61 E-4	7.61 E-4	
	1-4 days	4.84 E-4	4.84 E-4	4.84 E-4	4.84 E-4	
	4-30 days	2.13 E-4	2.13 E-4	2.13 E-4	2.13 E-4	
	SBGTS					
	0-2 hr	7.00 E-4	7.00 E-4	7.00 E-4	7.00 E-4	
	2-8 hr	6.45 E-6	6.45 E-6	6.45 E-6	6.45 E-6	
	8-24 hr	3.81 E-6	3.81 E-6	3.81 E-6	3.81 E-6	
	1-4 days	2.42 E-6	2.42 E-6	2.42 E-6	2.42 E-6	
	4-30 days	1.07 E-6	1.07 E-6	1.07 E-6	1.07 E-6	

Table 1

Assumptions and Inputs Used in Control Room Habitability Radiological Analyses

	• Assumption/Input	• Current Dresden Value	• Revised Dresden Value	• Current Quad Cities Value	• Revised Quad Cities Value	• Comments on Revised Value
III.	**Data and Assumptions for Control Room**					
A.	Volume of Control Room Emergency Zone, cu ft	132,000	81,000	184,000	184,000	Revised Dresden value of based on walled-off Unit 1 portion of control room and exclusion of computer room to reduce potential infiltration
B.	Volume of Control Room Proper, cu ft	104,000	64,000	58,000	58,000	Dresden value is based on walled-off Unit 1 portion of control room.
C.	Control Room Normal Intake Flow, cfm Control Room Emergency Intake Flow, cfm	2,000 2,000	2,200 (2000+10% margin) 1,620 cfm (2,000 minus 10% tolerance, minus 10% margin)	2,000 2000	2,200 (2000+10% margin) 1,620 cfm (2,000 minus 10% tolerance, minus 10% margin)	10% Tech Spec tolerance for max dose under emergency flow (during pressurization); no change to normal flow
D.	CR Intake Charcoal Adsorption Efficiencies for Iodines: Organic, % Elemental, % Particulate, %	 99 99 99	 99 99 99	 99 99 99	 99 99 99	Assumption consistent with Reg. Guide 1.52 and T.S.
E.	Time following start of the DBA (T=0) at which the normal intake is isolated and use of the Air Filtration Unit is initiated, minutes	40	40	110	40*	*Reduced to further lower operator dose, and becomes consistent with Dresden design basis
F.	Unfiltered inleakage, scfm	263	263 400* 500* 1000*	260	260 400* 500* 1000*	*Possible future use
G.	CR Cleanup Recirculation Flowrate, scfm	0	0	0	0	No credit taken for recirculation cleanup

Table 1
Assumptions and Inputs Used in Control Room Habitability Radiological Analyses

• Assumption/Input	• Current Dresden Value	• Revised Dresden Value	• Current Quad Cities Value	• Revised Quad Cities Value	• Comments on Revised Value
H. Occupancy Factors					Per SRP 6.4
0 to 1 day	1.0	1.0	1.0	1.0	
1 to 4 day	0.6	0.6	0.6	0.6	
4 to 30 day	0.4	0.4	0.4	0.4	
I. Effective X/Qs, sec/m^3 includes occupancy factor					Calculated, X/Q times occupancy factor
<u>Bypass</u>					
0-2 hr					
2-8 hr	1.29 E-3	1.29 E-3	1.29 E-3	1.29 E-3	
8-24 hr	1.29 E-3	1.29 E-3	1.29 E-3	1.29 E-3	
1-4 days	7.61 E-4	7.61 E-4	7.61 E-4	7.61 E-4	
4-30 days	2.90 E-4	2.90 E-4	2.90 E-4	2.90 E-4	
<u>SBGTS</u>	8.52 E-5	8.52 E-5	8.52 E-5	8.52 E-5	
0-2 hr					
2-8 hr	7.00 E-4	7.00 E-4	7.00 E-4	7.00 E-4	
8-24 hr	6.45 E-6	6.45 E-6	6.45 E-6	6.45 E-6	
1-4 days	3.81 E-6	3.81 E-6	3.81 E-6	3.81 E-6	
4-30 days	1.45 E-6	1.45 E-6	1.45 E-6	1.45 E-6	
	4.28 E-7	4.28 E-7	4.28 E-7	4.28 E-7	
J. Dose Conversion Factors	TID DCFs	ICRP 30 DCFs	TID DCFs	ICRP 30 DCFs	

Table 1
Assumptions and Inputs Used in Control Room Habitability Radiological Analyses

	• Assumption/Input	• Current Dresden Value	• Revised Dresden Value	• Current Quad Cities Value	• Revised Quad Cities Value	• Comments on Revised Value
IV.	**Data and Assumptions for Main Steam Line Break (if different from above)**					
A.	Quantity of Reactor Coolant Released	Not Used	66,000 lbs	Not Used	100,000 lbs	Based on current UFSAR analyses
B.	Quantity of Reactor Coolant and Fission Products that flash to Steam	Not Used	45,000 lbs	Not Used	55,000 lbs.	Based on current UFSAR analyses
C.	Specific Activity	Not Used	0.2 and 4.0 μCi/gm	Not Used	0.2 and 4.0 μCi/gm	Technical Specification Limit measured using ICRP 2
D.	Analysis Methodology		Puff Release - Uniform Cloud	Not Used	Puff Release - Uniform Cloud	The steam release analyzed as a puff release of uniform concentration that migrates across the CR intake at a rate of 1 meter per second
E.	Diameter of Cloud	Not Used	129.7 ft	Not Used	141.1 ft.	Calculated
F.	Duration of Cloud exposure at CR Intake	Not Used	40 seconds	Not Used	43 seconds	1 meter per second wind speed
G.	Rate at which Activity enters Control Room	Not Used	2,200 cfm	Not Used	2,200	Activity enters control room via normal outside air intake (plus 10% for margin). No credit for operation of filter unit. No credit at Quad Cities for Auto-isolation of intake.

TABLE 2
Control Room LOCA Dose Calculation Results
Dresden Units 2 & 3

The original design basis was rerun to ensure proper AXIDENT code function and inputs. The results are identical to those in NUS Calc. 546Y-M-09. The results reported in 546Y-M-09 were rounded to 23.1 rem for the SBGTS path and 6.22 rem for the MSIV path for a total dose of 29.33 rem. The results of this run are as follows:

Original Design Basis

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	5.69
Filtered Mode	17.40
MSIV Contribution	
Unfiltered Mode	4.71E-2
Filtered Mode	6.17
Total	29.31

The CR dose from the revised design basis analysis using the updated methodologies are as follows:

Revised Design Basis

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.45
Filtered Mode	2.81
MSIV Contribution	
Unfiltered Mode	9.90E-3
Filtered Mode	5.58
ESF Contribution	
Unfiltered Mode	2.53E-2
Filtered Mode	5.07E-2
Total	9.93

TABLE 2
Control Room LOCA Dose Calculation Results
Dresden Units 2 & 3

The whole body and beta doses for the original design basis case are identical to those previously calculated in the original dose analysis, NUS Calculation 546Y-M-06/S1, and NUS Calculation 546Y-M-09. While the whole body and beta doses for the revised design basis are slightly higher than the original design basis values, they are well within the 5 rem and 30 rem regulatory limits.

Original Design Basis

	Whole Body (rem)	Beta (rem)
SBGTS Contribution		
CR Activity	6.69E-2	1.33
Plume Shine	1.98E-2	
MSIV Contribution		
CR Activity	1.44E-2	4.70E-1
Plume Shine	2.03E-3	
Facility Shine	1.01E-1	
Total	2.04E-1	1.80

Revised Design Basis

	Whole Body (rem)	Beta (rem)
SBGTS Contribution		
CR Activity	1.56E-1	3.33
Plume Shine	1.98E-2	
MSIV Contribution		
CR Activity	1.32E-2	5.13E-1
Plume Shine	2.03E-3	
ESF Contribution	6.36E-4	1.36E-2
Facility Shine	1.01E-1	
Total	2.93E-1	3.86

TABLE 2
Control Room LOCA Dose Calculation Results
Dresden Units 2 & 3

The control room operator thyroid dose is reduced from 29.3 rem to 9.93 rem using the revised methodologies with 263 scfm infiltration. The calculated control room dose for the cases with higher control room infiltration rates are as follows:

Revised Design Basis w/400 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.43
Filtered Mode	3.90
MSIV Contribution	
Unfiltered Mode	9.77E-3
Filtered Mode	7.75
ESF Contribution	
Unfiltered Mode	2.49E-2
Filtered Mode	7.05E-2
Total	13.2

Revised Design Basis w/500 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.41
Filtered Mode	4.62
MSIV Contribution	
Unfiltered Mode	9.68E-3
Filtered Mode	9.17
ESF Contribution	
Unfiltered Mode	2.47E-2
Filtered Mode	8.34E-2
Total	15.3

Revised Design Basis w/1000 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.36
Filtered Mode	7.37
MSIV Contribution	
Unfiltered Mode	9.35E-3
Filtered Mode	14.6
ESF Contribution	
Unfiltered Mode	2.38E-2
Filtered Mode	1.33E-1
Total	23.5

TABLE 2
Control Room LOCA Dose Calculation Results
Dresden Units 2 & 3

With an alternate MSIV leak rate of 120 scfh instead of 46 scfh, the following calculated control room dose for the full range of control room infiltration rates (263, 400, 500, and 1000 cfm) are higher.

Revised Design Basis w/120 scfh MSIV and 263 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.45
Filtered Mode	2.81
MSIV Contribution	
Unfiltered Mode	2.58E-2
Filtered Mode	14.3
ESF Contribution	
Unfiltered Mode	2.53E-2
Filtered Mode	5.07E-2
Total	18.7

Revised Design Basis w/120 scfh MSIV and 400 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.43
Filtered Mode	3.90
MSIV Contribution	
Unfiltered Mode	2.55E-2
Filtered Mode	19.9
ESF Contribution	
Unfiltered Mode	2.49E-2
Filtered Mode	7.05E-2
Total	25.4

Revised Design Basis w/120 scfh MSIV and 500 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.41
Filtered Mode	4.62
MSIV Contribution	
Unfiltered Mode	2.53E-2
Filtered Mode	23.5
ESF Contribution	
Unfiltered Mode	2.47E-2
Filtered Mode	8.34E-2
Total	29.7

TABLE 2
Control Room LOCA Dose Calculation Results
Dresden Units 2 & 3

Revised Design Basis w/120 scfh MSIV and 1000 cfm Infiltration

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.36
Filtered Mode	7.37
MSIV Contribution	
Unfiltered Mode	2.44E-2
Filtered Mode	37.5
ESF Contribution	
Unfiltered Mode	2.38E-2
Filtered Mode	1.33E-1
Total	46.4

TABLE 2A
Control Room LOCA Dose Calculation Results
Siemens Fuel
Dresden Unit 2 & 3

The calculated control room dose from the Siemens fuel with a 20,000 Mwd/MTU burnup are 5.7% higher than with the TID fuel. The maximum depletion with 60,000 Mwd/MTU burnup yields a 14.8% higher dose.

Revised Design Basis w/20,000 Mwd/MTU Siemens Fuel

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.49
Filtered Mode	2.94
MSIV Contribution	
Unfiltered Mode	1.02E-2
Filtered Mode	5.96
ESF Contribution	
Unfiltered Mode	2.61E-2
Filtered Mode	5.31E-2
Total	10.5

Revised Design Basis w/60,000 Mwd/MTU Siemens Fuel

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	1.59
Filtered Mode	3.17
MSIV Contribution	
Unfiltered Mode	1.09E-2
Filtered Mode	6.54
ESF Contribution	
Unfiltered Mode	2.77E-2
Filtered Mode	5.73E-2
Total	11.4

Table 4
Control Room MSLB Dose Calculation Results
Dresden Units 2 & 3 and Quad Cities Units 1 & 2

Dresden:

Coolant Activity (µCi/g)	CR Infiltration (cfm)	Thyroid Dose (rem)
0.2	2200	1.21
4.0	2200	24.2

Quad Cities:

Coolant Activity (µCi/g)	CR Infiltration (cfm)	Thryoid Dose (rem)
0.2	2200	1.22
4.0	2200	24.4

TABLE 5
Offsite LOCA Dose Calculation Results
Dresden Units 2 & 3 and Quad Cities Units 1 & 2

DRESDEN

EAB Dose Comparison

	Thyroid (rem)	Wholebody (rem)
Original SER	185	8
Revised methodology (no mixing)	6.36	5.91
Revised Methodology (mixing)	0.47	0.30

LPZ Dose Contribution

	Thyroid (rem)	Wholebody (rem)
Original SER	90	2
Revised methodology (no mixing)	5.64	1.39
Revised methodology (mixing)	3.64	0.32

QUAD CITIES

EAB Dose Comparison

	Thyroid (rem)	Wholebody (rem)
Original SER	150	6
Revised methodology (no mixing)	6.68	6.7
Revised methodology (mixing)	0.46	0.26

LPZ Dose Contribution

	Thyroid (rem)	Wholebody (rem)
Original SER	108	3
Revised methodology (no mixing)	5.84	1.46
Revised methodology (mixing)	3.71	0.33

TABLE 3A
Control Room LOCA Dose Calculation
Siemens Fuel
Quad Cities Units 1 & 2

The calculated control room dose from the Siemens fuel with a 20,000 Mwd/MTU burnup are approximately 6% higher than with the TID fuel. The maximum depletion with 60,000 Mwd/MTU burnup yields a 15.5% higher dose.

Revised Design Basis w/20,000 Mwd/MTU Siemens Fuel

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	8.39E-1
Filtered Mode	1.64
MSIV Contribution	
Unfiltered Mode	1.11E-2
Filtered Mode	6.21
ESF Contribution	
Unfiltered Mode	2.54E-2
Filtered Mode	5.06E-2
Total	8.78

Revised Design Basis w/60,000 Mwd/MTU Siemens Fuel

Leakage Path	Thyroid Dose (rem)
SBGTS Contribution	
Unfiltered Mode	8.96E-1
Filtered Mode	1.77
MSIV Contribution	
Unfiltered Mode	1.18E-2
Filtered Mode	6.80
ESF Contribution	
Unfiltered Mode	2.72E-2
Filtered Mode	5.47E-2
Total	9.56