FENOC

FirstEnergy Nuclear Operating Company

Perry Nuclear Power Plant
10 Center Road
Perry, Ohio 44081

William R. Kanda
Vice President - Nuclear

440-280-5579
Fax. 440-280-8029

December 9, 2002
PY-CEI/NRR-2674L

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Perry Nuclear Power Plant
Docket No. 50-440
License Amendment Request Pursuant to 10 CFR 50.90: Alternative Source Term Update of the Design Basis Analyses for the Fuel Handling Accident

Ladies and Gentlemen:

A license amendment is requested for the Perry Nuclear Power Plant (PNPP). The requested change utilizes Alternative Source Term radiological calculations to update the design basis analysis for the Fuel Handling Accident. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", was utilized in the development of this application.

Approval is requested by March 14, 2003, to support preparations for the ninth refueling outage. This application is considered a cost beneficial licensing change due to anticipated cost savings on outage duration, due to redefinition of the term "recently irradiated fuel." If you have questions or require additional information, please contact Mr. Vernon K. Higaki, Manager - Regulatory Affairs, at (440) 280-5294.

Very truly yours,

Enclosures:

1. Notarized Affidavit
2. Evaluation of the changes, including a Summary, Description of the Changes, Background, Technical Analysis, Regulatory Analysis/Commitments, and Environmental Consideration
3. Significant Hazards Consideration
4. Dose Calculation entitled "Fuel Handling Accident Using Alternative Source Term"
5. Information copy of proposed Updated Safety Analysis Report (USAR) changes (mark-up)

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III
State of Ohio

A001

I, William R. Kanda, hereby affirm that (1) I am Vice President - Perry, of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification as the duly authorized agent for The Cleveland Electric Illuminating Company, Toledo Edison Company, Ohio Edison Company, and Pennsylvania Power Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

William R. Kanda

Subscribed to and affirmed before me, the 9th day of December, 2003

Brenda Alvord
Notary Public
Lake Co.
State of Oh

My commission expires on 8-15-06.

Summary

The amendment proposed to the Perry Nuclear Power Plant (PNPP) license is based on a new dose analysis for the design basis Fuel Handling Accident, using an Alternative Source Term (AST). Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", was utilized in the development of this application.

The analysis assumes the event occurs after only one day (24 hours) of radiological decay, rather than seven days as was assumed in License Amendment 102 (issued in March 1999). Even with this 24-hour assumption, the doses from such an event remain within regulatory acceptance limits.

Description of the Changes

A license amendment is requested based on 10 CFR 50.67.b(1) and the Nuclear Regulatory Commission (NRC) position in Section 1.1.1 of Regulatory Guide 1.183 that an initial AST implementation should be approved per 10 CFR 50.90. This is the initial use of an alternative source term for the Fuel Handling Accident design basis analysis at PNPP. Therefore, the calculation is provided in Enclosure 4 to this submittal for NRC review and approval of a license amendment per 10 CFR 50.90.

Background

In a Federal Register Notice dated December 23, 1999, the Nuclear Regulatory Commission (NRC) published a new regulation, 10 CFR 50.67, providing a mechanism for licensed power reactors to replace the traditional accident source term used in design basis accident analyses with Alternative Source Terms (ASTs). Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", dated July 2000. 10 CFR 50.67(b)(1) states that licensees who seek to revise their current accident source term in design basis radiological consequence analyses should apply for a license amendment under 10 CFR 50.90.

Two previous PNPP license amendments have laid the groundwork for the current license amendment request. License Amendment 102, issued in March 1999, introduced the concept that Shutdown Safety administrative controls can be utilized during fuel handling once the dose calculations demonstrate that regulatory limits for the Fuel Handling Accident can be met without credit for filtration systems and the Containment/Fuel Handling Buildings. License Amendment 103, also issued in March 1999, involved a pilot plant application of an alternative source term for a design basis Loss Of Coolant Accident (LOCA). Subsequent to March of 1999, significant consideration was given to the characteristics of the AST for a Fuel Handling Accident (FHA), which is different from a LOCA. The results of these considerations were published by the NRC in Regulatory Guide 1.183, which is the primary regulatory basis document for the currently proposed change. This current request therefore applies the AST characteristics of a Fuel Handling Accident to the PNPP radiological calculations for handling fuel that has been subcritical for at least 24 hours.

Technical Analysis

The following table summarizes conformance to Regulatory Guide 1.183, to ensure the guidance is adequately addressed. This supplements the actual calculation, which is included as Enclosure 4.

Regulatory Guide 1.183 Guidance	Degree of Conformance
Section 1 Implementation of AST *1.1.1 Safety Margins* "The proposed uses of an AST and the associated	Sufficient safety margins are maintained with the

Regulatory Guide 1.183 Guidance	Degree of Conformance
proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. The safety margins are products of specific values and limits contained in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times. Changes, or the net effects of multiple changes, that result in a reduction in safety margins may require prior NRC approval. Once the initial AST implementation has been approved by the staff and has become part of the facility design basis, the licensee may use 10 CFR 50.59 and its supporting guidance in assessing safety margins related to subsequent facility modifications and changes to procedures."	Alternative Source Term analyses. There are a number of conservatisms in the calculations, which account for analysis uncertainties. The primary uncertainties in the calculation consist of the inventory released from the fuel, the scrubbing of the nuclides from the water pool over the fuel, the efficiency of filtration provided by plant structures and systems, and dispersion of the release as it travels away from the plant. The inventory available in the fuel is determined using an NRC accepted code (see Section 3.1 below) which is conservative enough to address uncertainties in the inventory, and in the radiological decay process. The fraction of that inventory which is available in the gap of the fuel rods is assumed to be the same as provided in Regulatory Guide 1.183 (see Section 3.2 below), which addresses uncertainties in the gap fraction. The uncertainties of the scrubbing provided by the water is addressed by using the overall decontamination factor (DF) of 200 documented in Regulatory Guide 1.183. This is a conservative value. Several plants that have submitted Alternative Source Term analyses have shown that the overall DF provided by the water over the fuel is actually greater than the 200 value. The requirements for water coverage over the fuel in the Technical Specifications remain unchanged by this proposal. The uncertainties in the efficiency of filtration systems to treat the release are addressed by assuming there are no Containment or Fuel Handling Buildings or ventilation/filtration systems present, and the release from the pool to the environment is an instantaneous, undiluted, and unfiltered release. The release is then dispersed by Chi/Q values previously approved by the NRC, which were considered to adequately address uncertainties in the actual dispersion of the release. The dose calculation results remain below the limits of 10 CFR 50.67. Table 7 of Enclosure 4 presents the results of the base calculation (Table 8 presents sensitivities), along with the applicable dose limits for the Control Room, Exclusion Area Boundary (EAB), and the Low Population Zone (LPZ): **TABLE 7 RESULTS** Control Room / EAB / LPZ RADTRAD Results (rem): 1.03 / 1.44 / 0.161 Regulatory limit (rem): 5 / 6.3 / 6.3

Regulatory Guide 1.183 Guidance	Degree of Conformance
1.1.2 Defense in Depth "The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties. In all cases, compliance with the General Design Criteria in Appendix A to 10 CFR Part 50 is essential. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions. ... Proposed modifications that seek to downgrade or remove required engineered safeguards equipment should be evaluated to be sure that the modification does not invalidate assumptions made in facility PRAs and does not adversely impact the facility's severe accident management program."	Adequate defense in depth is maintained through the use of Technical Specification controls over buildings and filtration systems when fuel being handled is "recently irradiated fuel", and using the controls described and approved in License Amendment 102, when fuel is not recently irradiated. These are in addition to the natural defenses of radiological decay over time (which reduces the magnitude of any release) and the scrubbing effect of the water pool over the fuel (the controls over water level are not being changed as a result of this amendment). As noted above, the radiological calculations show that Fuel Handling Accident doses remain within regulatory acceptance limits. Fuel Handling Accidents are not modeled by the PNPP Probabilistic Safety Analysis (PSA). However, risk is assessed and managed in accordance with plant procedure, which uses defense in depth criteria and other considerations as specified in the controls put in place by License Amendment 102. Similarly, the Severe Accident Management (SAM) Program would not be affected by the proposed change, based on the maintenance of the controls put in place per Amendment 102 over buildings and filtration systems during plant shutdowns.
1.1.3 Integrity of Facility Design Basis "...Although a complete re-assessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all design analyses would generally not be necessary. Regulatory Position 1.3 of this guide provides guidance on which analyses need updating as part of the AST implementation submittal and which may need updating in the future as additional modifications are performed. This approach would create two tiers of analyses, those based on the previous source term and those based on an AST. ... In either case, the facility design bases should clearly indicate that the source term assumptions and radiological criteria in these affected analyses have been superseded and that future revisions of these analyses, if any, will use the updated approved assumptions and criteria. ..."	See further discussion under Regulatory Position 1.3 below, and see the example markups of the Updated Safety Analysis Report (USAR) provided for information in Enclosure 5. These note that this application is considered to be a selective application of the AST. The USAR markups note that the source term assumptions and radiological criteria in the previous Fuel Handling Accident analyses have been superceded by the new analyses, and future revisions of Fuel Handling Accident analyses will use the updated source term assumptions and radiological criteria.
1.1.4 Emergency Preparedness Applications "...The AST is not representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness. Therefore, the AST is insufficient by itself as a basis	No relief is being requested from emergency planning provisions. Procedures are already in place for responding to a

Regulatory Guide 1.183 Guidance	Degree of Conformance
for requesting relief from the emergency preparedness requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50. This guidance does not, however, preclude the appropriate use of the insights of the AST in establishing emergency response procedures such as those associated with emergency dose projections, protective measures, and severe accident management guides."	fuel handling accident as a result of License Amendment 102.
1.2.1 Full Implementation	This application is not considered to be a full implementation. See Section 1.2.2 below.
1.2.2 Selective Implementation "Selective implementation is a modification of the facility design basis that (1) is based on one or more of the characteristics of the AST or (2) entails re-evaluation of a limited subset of the design basis radiological analyses. The NRC staff will allow licensees flexibility in technically justified selective implementations provided a clear, logical, and consistent design basis is maintained. An example of an application of selective implementation would be ... a request to remove the charcoal filter media from the spent fuel building ventilation exhaust. For the latter, the licensee may only need to re-analyze DBAs that credited the iodine removal by the charcoal media. Additional analysis guidance is provided in Regulatory Position 1.3 of this guide.	This application is a selective implementation. It entails re-evaluation of a limited subset of the design basis radiological analyses, specifically the Fuel Handling Accident. The only DBA that needs to be reanalyzed for this amendment is the Fuel Handling Accident. The description of how the design basis for this event is being maintained is included in the example USAR markups in Enclosure 5.
NRC approval for the AST (and the TEDE dose criterion) will be limited to the particular selective implementation proposed by the licensee. The licensee would be able to make subsequent modifications to the facility and changes to procedures based on the selected AST characteristics incorporated into the design basis under the provisions of 10 CFR 50.59. However, use of other characteristics of an AST or use of TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics, would require prior staff approval under 10 CFR 50.67..."	It is understood that since this is a selective application of the AST for the Fuel Handling Accident, NRC approval will be limited to this event. Use of AST to change the design basis for other events such as the Control Rod Drop Accident or the Main Steam Line Break Outside Containment, or changes to the approved AST characteristics, would require prior staff approval under 10 CFR 50.67.
1.3.1 Design Basis Radiological Analyses "There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of design basis accidents. These requirements include, but are not limited to, the following. • Environmental Qualification of Equipment (10 CFR 50.49) • Control Room Habitability (GDC-19 of Appendix A to 10 CFR Part 50) • Emergency Response Facility Habitability (Paragraph IV.E.8 of Appendix E to 10 CFR Part 50)	**10 CFR 50.49 Environmental Qualification of Equipment** – No credit is taken for filtration system OPERABILITY (or OPERABILITY of any other system) in the design basis calculations for the Fuel Handling Accident after 24 hours of radiological decay. Therefore, there is not a concern that some aspect of the alternative source term could make such systems unable to perform a "credited" safety function. **GDC 19 Control Room Habitability** – For a Fuel Handling Accident, a design basis dose calculation for the Control Room was performed assuming 24 hours of radiological decay. The base calculation

Regulatory Guide 1.183 Guidance	Degree of Conformance
♦ Alternative Source Term (10 CFR 50.67) ♦ Environmental Reports (10 CFR Part 51) ♦ Facility Siting (10 CFR 100.11) There may be additional applications of the accident source term identified in the Technical Specification bases and in various licensee commitments. These include, but are not limited to, the following from Reference 2, NUREG-0737. ♦ Post-Accident Access Shielding (NUREG-0737, II.B.2) ♦ Post-Accident Sampling Capability (NUREG-0737, II.B.3) ♦ Accident Monitoring Instrumentation (NUREG-0737, II.F.1) ♦ Leakage Control (NUREG-0737, III.D.1.1) ♦ Emergency Response Facilities (NUREG-0737, III.A.1.2) ♦ Control Room Habitability (NUREG-0737, III.D.3.4)."	showed that doses remained less than the 5 rem Total Effective Dose Equivalent (TEDE) limit in GDC-19 and 10 CFR 50.67, even assuming no credit for Containment or FHB or filtration systems . **10 CFR 50 Appendix E Emergency Response Facility Habitability** – The proposed changes do not result in changes to Emergency Response Facility Habitability. 10 CFR 50 Appendix E does not contain habitability criteria, however NUREG-0737 Supplement 1 does. The only facility with a specific dose criterion is the Technical Support Center (TSC). The dose limit in Supplement 1 for this facility is 5 rem whole body, or its equivalent. The "or equivalent" for this evaluation is considered to be 5 rem TEDE. Although the TSC has essentially no response function for a Fuel Handling Accident, a scoping study for the TSC was performed. The ventilation intakes for the TSC are farther away from the containment structure and from ventilation system release points than the Control Room intakes, and the TSC intake is at a lower elevation by more than 60 feet. Since the dispersion of a plume for an intake at a greater distance and lower elevation would be correspondingly better, the scoping evaluation concluded that the 5 rem TEDE limit would be met for the TSC as well. The regulatory guidance does not include specific dose limits for Emergency Operations Facility (EOF) and backup EOF habitability. For the same reasons as discussed for the TSC, these facilities are also considered to not be adversely affected as a result of this change in the source term assumptions. **10 CFR 50.67 Accident Source Term** – The acceptance criteria of 10 CFR 50.67 and the attributes of an acceptable alternative source term as described in Regulatory Guide 1.183 are being utilized in this application. **10 CFR Part 51 Environmental Protection Regulations** – See the section of this letter entitled "Environmental Consideration" below. **10 CFR 100.11 Facility Siting** – As noted in Footnote 5 of Reg. Guide 1.183, the dose guidelines of 10 CFR 100.11 are superceded by 10 CFR 50.67 for applications implementing an alternative source term such as this. **NUREG-0737 Item II.B.2 Post-Accident Access Shielding** – There are no design basis actions credited outside the Control Room for a Fuel Handling Accident. TSC access/dose was addressed above. **NUREG-0737 Item II.B.3 Post-Accident Sampling Capability** – No post-accident sampling inside the Containment is required for a Fuel Handling

Regulatory Guide 1.183 Guidance	Degree of Conformance
	Accident. **Accident Monitoring Instrumentation (NUREG-0737, II.F.1)** - No post-accident monitors are required to respond to a Fuel Handling Accident. **NUREG-0737 Item III.D.1.1 Leakage Control** – No post-accident leakage control is required for a Fuel Handling Accident. **NUREG-0737 Item III.A.1.2 Emergency Response Facilities** – Item III.A.1.2 is unaffected, since no dose protection or habitability guidance is included in this TMI item. See discussions above on Emergency Response Facilities. **NUREG-0737, Item III.D.3.4 Control Room Habitability** – Control Room habitability was analyzed and determined to be acceptable, by meeting the radiological dose limits of 10 CFR 50.67. The proposed amendment does not affect protection from toxic gases. No additional applications of the accident source term for a Fuel Handling Accident were identified in the Technical Specification Bases or in licensee commitments.
1.3.2 Re-Analysis Guidance "Any implementation of an AST, full or selective, and any associated facility modification should be supported by evaluations of all significant radiological and non-radiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above as well as any other facility-specific requirements. These impacts may be due to (1) the associated facility modifications or (2) the differences in the AST characteristics.	The change consists of a redefinition of the term "recently irradiated fuel" to be consistent with the new dose calculations. Compliance with various regulations and commitments are addressed in this Table. Technical Specification controls exist if "recently irradiated fuel" is handled; the controls put in place per Amendment 102 continue to apply during other periods of fuel handling.
The scope and extent of the re-evaluation will necessarily be a function of the specific proposed facility modification and whether a full or selective implementation is being pursued. The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid. Generic analyses, such as those performed by owner groups or vendor topical reports, may be used provided the licensee justifies the applicability of the generic conclusions to the	The design basis FHA calculation has been updated and is included in Enclosure 4 for NRC review. This selective implementation is solely for the Fuel Handling Accident. Other design basis calculations were determined to not be affected by this proposed license amendment. Example USAR markups are also provided for information in Enclosure 5.

Regulatory Guide 1.183 Guidance	Degree of Conformance
specific facility and implementation. Sensitivity analyses, discussed below, may also be an option. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed. The license amendment request should describe the licensee's re-analysis effort and provide statements regarding the acceptability of the proposed implementation, including modifications, against each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide. ..."	In the calculation, all affected assumptions and inputs were updated to address AST and TEDE, and all selected characteristics of the AST and the TEDE criteria are addressed. Statements regarding the acceptability of the proposed amendment against each of the applicable items identified in Regulatory Position 1.3.1 of the Reg. Guide were provided above. The above discussion addressed radiological impact of the proposed change. Since there are no physical design modifications being made in conjunction with this proposal, there are also no non-radiological impacts as a result of the proposed change.
1.3.3 Use of Sensitivity or Scoping Analyses "It may be possible to demonstrate by sensitivity or scoping evaluations that existing analyses have sufficient margin and need not be recalculated. As used in this guide, a sensitivity analysis is an evaluation that considers how the overall results vary as an input parameter (in this case, AST characteristics) is varied. A scoping analysis is a brief evaluation that uses conservative, simple methods to show that the results of the analysis bound those obtainable from a more complete treatment. Sensitivity analyses are particularly applicable to suites of calculations that address diverse components or plant areas but are otherwise largely based on generic assumptions and inputs. Such cases might include post-accident vital area access dose calculations, shielding calculations, and equipment environmental qualification (integrated dose). It may be possible to identify a bounding case, re-analyze that case, and use the results to draw conclusions regarding the remainder of the analyses. It may also be possible to show that for some analyses the whole body and thyroid doses determined with the previous source term would bound the TEDE obtained using the AST. Where present, arbitrary "designer margins" may be adequate to bound any impact of the AST and TEDE criteria. If sensitivity or scoping analyses are used, the license amendment request should include a discussion of the analyses performed and the conclusions drawn. Scoping or sensitivity analyses should not constitute a significant part of the evaluations for the design basis exclusion area boundary (EAB), low population zone (LPZ), or	No sensitivity evaluations that varied AST characteristics were performed. However, several sensitivity evaluations were performed which varied Control Room ventilation assumptions to show doses remained acceptable. In the base case, normal ventilation continues to operate throughout the event, which initially brings the undiluted, unfiltered source term directly into the Control Room without any isolation protection. This case takes no credit for the Control Room Area Radiation Monitor or the Emergency Recirculation (filtration) system. Two other calculation sensitivity cases were also run, which isolated the control room at the worst possible time, after the source term available at the intake is introduced into the Control Room. For these two cases, the isolation exists for 2 hours, and then was followed by either a subsequent re-initiation of the normal intake flow, or the use of the filtration system. Each case produced an acceptable result, showing that following a Fuel Handling Accident, the operators have flexibility on how to operate their ventilation systems without exceeding the radiological acceptance criteria. The base case shows that filtration systems are not required. The two sensitivity cases show that ventilation/filtration systems can be effectively used to reduce doses to the Control Room operators in the event that the radiation monitor isolates the Control Room intake at the worst possible time after available activity is taken into the Control Room. A scoping evaluation is also used to show that the

Regulatory Guide 1.183 Guidance	Degree of Conformance
control room dose."	TSC doses would be lower than the Control Room doses since the TSC inlet is farther away from the plant vents and the Containment itself than the Control Room inlets are, and is lower on the buildings by more than 60 feet, so the dispersion factors would be better than the previously NRC-approved dispersion factors for the Control Room intakes.
1.3.4 Updating Analyses Following Implementation "Full implementation of the AST replaces the previous accident source term with the approved AST and the TEDE criteria for all design basis radiological analyses. ... Since [for a full implementation] the AST and the TEDE criteria are part of the approved design basis for the facility, use of the AST and TEDE criteria in new applications at the facility do not constitute a change in analysis methodology that would require NRC approval. This guidance is also applicable to selective implementations to the extent that the affected analyses are within the scope of the approved implementation as described in the facility design basis. In these cases, the characteristics of the AST and TEDE criteria identified in the facility design basis need to be considered in updating the analyses. Use of other characteristics of the AST or TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics, requires prior NRC staff approval under 10 CFR 50.67."	This is a selective implementation rather than a full implementation. Since the USAR discussions of the Fuel Handling Accident will include the AST and TEDE criteria (see Enclosure 5), future updates to Fuel Handling Accident calculations will continue to use the characteristics of the AST and TEDE under the provisions of 10 CFR 50.59.
1.3.5 Equipment Environmental Qualification "Current environmental qualification (EQ) analyses may be impacted by a proposed plant modification associated with the AST implementation. The EQ analyses that have assumptions or inputs affected by the plant modification should be updated to address these impacts. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. ..."	There are no increased EQ requirements as a result of this proposed amendment. Further details on EQ are provided in Section 6. The cesium issue discussed in this section of the Regulatory Guide is associated with a LOCA and is unrelated to a Fuel Handling Accident.
1.4 Risk Implications "The use of an AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents. The AST has no direct effect on the probability of the accident. Use of an AST alone cannot increase the core damage frequency (CDF) or the large early release frequency (LERF). However, facility modifications made possible by the AST could have an impact on risk. If the proposed implementation of the AST involves changes to the facility design that would invalidate assumptions made in the facility's PRA, the impact	As noted in Section 1.1.2, the PNPP PSA model is not affected by the proposed amendment.

Regulatory Guide 1.183 Guidance	Degree of Conformance
on the existing PRAs should be evaluated. Consideration should be given to the risk impact of proposed implementations that seek to remove or downgrade the performance of previously required engineered safeguards equipment on the basis of the reduced postulated doses. The NRC staff may request risk information if there is a reason to question adequate protection of public health and safety. The licensee may elect to use risk insights in support of proposed changes to the design basis that are not addressed in currently approved NRC staff positions. ..."	Although risk insights are not being used to support this change, some risk insights were utilized in NRC approval of License Amendment 102, which remain applicable to this proposed change. --
1.5 Submittal Requirements " ... The NRC staff's finding that the amendment may be approved must be based on the licensee's analyses, since it is these analyses that will become part of the design basis of the facility. The amendment request should describe the licensee's analyses of the radiological and nonradiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. The staff recommends that licensees submit affected FSAR pages annotated with changes that reflect the revised analyses or submit the actual calculation documentation. If the licensee has used a current approved version of an NRC-sponsored computer code, the NRC staff review can be made more efficient if the licensee identifies the code used ...."	The dose analysis calculation is being provided for NRC review as Enclosure 4. Additional detail on how the NRC guidance in Regulatory Guide 1.183 is being met is provided in this table format. USAR pages, which include examples of the types of changes that will be made, are also included for information as Enclosure 5. The Code used in the analysis was RADTRAD 3.02, January 5, 2000.
1.6 FSAR Requirements " ... The regulations in 10 CFR 50.71(e) require that the FSAR be updated to include all changes made in the facility or procedures described in the FSAR.... The affected radiological analysis descriptions in the FSAR should be updated to reflect the replacement of the design basis source term by the AST. The analysis descriptions should contain sufficient detail to identify the methodologies used, significant assumptions and inputs, and numeric results. ... The descriptions of superseded analyses should be removed from the FSAR in the interest of maintaining a clear design basis.	USAR pages are provided for information in Enclosure 5, which provide examples of how the licensing basis will be revised as a result of this proposed amendment.
Section 2 Attributes Of An Acceptable AST "...Regulatory Position 3 of this guide identifies an AST that is acceptable to the NRC staff for use at operating power reactors. A substantial effort was expended by the NRC, its contractors, various national laboratories, peer reviewers, and others in performing severe accident research and in developing the source terms provided in NUREG-	This application uses the characteristics of the source term outlined in Regulatory Position 3 of Reg. Guide 1.183. Therefore the rest of Section 2 is considered to be not applicable, since no attempt is made to define different source term characteristics from those provided in the Reg. Guide.

Regulatory Guide 1.183 Guidance	Degree of Conformance
1465 (Ref. 5). However, future research may identify opportunities for changes in these source terms. The NRC staff will consider applications for an AST different from that identified in this guide."...	
Section 3 Accident Source Term *3.1 Fission Product Inventory* "The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). ...	General Electric (GE) used the computer code ORIGEN 2 to determine the core inventory for a Fuel Handling Accident. This input was originally developed to support the power uprate and 24-month operating cycle amendments (License Amendments 112 and 115). The core inventory provided by GE was performed in Curies per megawatt (Ci/MW). The inventory was adjusted by an additional 2% to account for evaluation uncertainty.
For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods. ...	The fission product inventory of each of the fuel rods was determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, a radial peaking factor of 2.0 was applied. This simulates that the rods in the bundle being dropped and the struck bundles would be the highest inventory rods in the core. The maximum core wide radial peaking factor of 2.0 is being added into the list of reload analysis parameters that must be re-verified each cycle. [See Commitment 1 at the end of this enclosure].
For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled."	For the Fuel Handling Accident analyses performed for this submittal, radioactive decay from the time of shutdown was modeled.
3.2 Release Fractions " ... For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor."	Table 3 fractions were applied to the fission product inventory determined as described above for the rods with the maximum core radial peaking factor. These fractions are 8% of I-131, 10% of Kr-85, 5% of the other Noble Gases, 5% of the other Halogens, and 12% of the Alkali Metals.
[An applicable footnote is linked to Table 3. Footnote 11 states "The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an	For footnote 11, which applies to Table 3, the provisions in the first sentence of the footnote are met at PNPP. The fuel in use at PNPP is NRC approved fuel, and the average exposure of the peak fuel rod is maintained below 62,000 MWD/MTU (= 62 GWD/MTU). Also, the maximum linear heat generation rate for the fuel that could exceed 54 GWD/MTU by the end of the cycle is

Regulatory Guide 1.183 Guidance	Degree of Conformance
alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. ..."]	maintained at or below 6.3 kw/ft (in other words, the higher burnup fuel is moved to lower power portions of the core such as the periphery). The burnup limit of 62 GWD/MTU on the average exposure of the peak rod, and the LHGR limit of 6.3 kw/ft peak rod average power for the higher burnup fuel (> 54 GWD/MTU), are both being added into the list of reload analysis parameters that must be re-verified each cycle. [See Commitment 2]
3.3 Timing of Release Phases "... For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage. ..."	For the Fuel Handling Accident, the release from the fuel gap is assumed to occur instantaneously with the impact of the fuel bundle.
3.4 Radionuclide Composition Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** Radionuclide Groups **Group** — **Elements** Noble Gases — Xe, Kr Halogens — I, Br Alkali Metals — Cs, Rb Tellurium Group — Te, Sb, Se, Ba, Sr Noble Metals — Ru, Rh, Pd, Mo, Tc, Co Lanthanides — La, Zr, Nd, Eu, Nb, Pm, Pr Sm, Y, Cm, Am Cerium — Ce, Pu, Np	This guidance is generic for all events. More specific guidance for a Fuel Handling Accident is provided in Appendix B to Reg. Guide 1.183. In summary, only the first three groups in this table are considered to be available in the gap for immediate release (the Noble Gases, the Halogens, and the Alkali Metals). However, the Alkali Metals (Cesium and Rubidium) are particulates that have an infinite decontamination factor (i.e., they are fully retained by the water in the fuel pool or reactor cavity). Twenty of the most significant Noble Gases and Halogens are used in the calculation for a Fuel Handling Accident (see Enclosure 4). The other nuclides in these groups were not included because their core activity was less than 1E-9 Ci/MWt, and were considered insignificant.
3.5 Chemical Form " ... The accident-specific appendices to this Regulatory Guide provide additional details."	Specific details on Chemical Form for Fuel Handling Accidents are in the Appendix B discussions below.
3.6 Fuel Damage in Non-LOCA DBAs " ... The amount of fuel damage caused by a FHA is addressed in Appendix B of this guide.	See the fuel pin failure discussion below for the Appendix B items.
Section 4 Dose Calculational Methodology "The NRC staff has determined that there is an implied synergy between the ASTs and total effective dose equivalent (TEDE) criteria, and between the TID-14844 source terms and the whole body and thyroid dose criteria, and therefore, they do not expect to allow the TEDE criteria to be used with TID-14844 calculated results. The guidance of this section applies to all dose calculations performed with an AST pursuant to 10 CFR 50.67."	The Total Effective Dose Equivalent (TEDE) criteria are utilized in this AST application, which is performed pursuant to 10 CFR 50.67.
4.1 Offsite Dose Consequences "...*4.1.1* ... TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides,	The TEDE dose calculations considered the radionuclides, including progeny from the decay of parent radionuclides, which are significant with regard to dose consequences and the released radioactivity. All the isotopes of bromine, iodine,

Regulatory Guide 1.183 Guidance	Degree of Conformance
including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity."	krypton, and xenon with core activity greater than 1E-9 Ci/MWt (a total of 20) and their daughters, i.e., an additional three isotopes of cesium and rubidium, were used.
4.1.2 The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	The conversion factors utilized for the CEDE inhalation component (of TEDE) were obtained from the 1989 printing of Federal Guidance Report 11.
4.1.3 For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second."	The recommended offsite Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) breathing rates were used, however, considering that the release would occur instantaneously, the effective breathing rate used was 3.5E-4 m^3/s.
4.1.4 The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	The conversion factors utilized for the DDE/EDE external component (of TEDE) were obtained from the MACCS2 computer code, which uses the 1993 version of Federal Guidance Report 12.
4.1.5 ... The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined ... by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted ... (see also Table 6).	The activity is conservatively assumed to be immediately released (a puff release rather than a 2-hour period). Atmospheric dispersion of radioactivity during transport was accounted for by using the PNPP dispersion factors (Chi/Q), but the release was transported to the EAB and the LPZ immediately, without delay or deposition on the ground. Therefore, it was not necessary to perform sliding sums. Table 6 of Reg. Guide 1.183 identifies the FHA analysis release duration as 2 hours. The puff release replaced that assumption,

Regulatory Guide 1.183 Guidance	Degree of Conformance
	and is considered conservative.
4.1.6 TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	The TEDE dose was determined for the most limiting receptor at the outer boundary of the LPZ. The results, and the 10 CFR 50.67 limits, are presented in Table 7 of Enclosure 4.
4.1.7 No correction should be made for depletion of the effluent plume by deposition on the ground."	No credit was taken in the calculations for deposition of the radionuclides on the ground.
4.2 Control Room Dose Consequences "...*4.2.1* The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: ◆ Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, ◆ Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, ◆ Radiation shine from the external radioactive plume released from the facility, ◆ Radiation shine from radioactive material in the reactor containment, ◆ Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	All the radioactivity released from the pool is assumed to be immediately transported outside of the Containment without dilution. Contamination of the Control Room atmosphere by the intake of the available radioactive material contained in the radioactive plume was modeled. Infiltration in addition to the 6600 cfm of unfiltered intake was not incorporated, since there are no in-plant pathways that can transport activity to within the Control Room as effectively as via the outside air intake (additional information is provided in Section 3.14.1.2 of the calculation in Enclosure 4). In the event that the Control Room intake isolates and the activity is trapped in the Control Room, assuming lesser quantities of infiltration is conservative, since subsequent inleakage would dilute/purge the trapped activity. Due to shielding of the Control Room, radiation shine from a Fuel Handling Accident is considered to be a negligible dose contributor. More details on the various assumptions for radiation sources and the shielding available to the Control Room is provided in the calculation, Section 3.14.
4.2.2 The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	The radioactive material releases and radiation levels used in the Control Room dose analysis were determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values. Control Room Chi/Q values were utilized.
4.2.3 The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	The RADTRAD computer code was used to model transport of radioactive material into and through the Control Room. This modeling provides suitably conservative estimates of the exposure to Control Room personnel.
4.2.4 Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or	The base calculation takes no credit for Control Room engineered safety features or isolations, i.e., no credit for the Control Room radiation monitor that can isolate the intake, or for any filtration on the

Regulatory Guide 1.183 Guidance	Degree of Conformance
intake or recirculation filtration. ... In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.	intake flows. Since no credit was taken for isolation of the intake by the radiation monitor, the issue of whether this monitor might be delayed in responding to the radiation is not a concern. Sensitivity studies were performed to examine the flexibility the Control Room operators have in using ventilation, to ensure there were no dose outliers. The studies evaluated what steps could be taken even if the radiation monitor was to isolate the intake at the worst possible time (after all of the available activity from the plume had been introduced into the Control Room). The sensitivity studies showed that even if the operators take two hours to take action, they can then either purge or use ventilation filters to remove the activity, and neither method resulted in excessive doses. Procedural guidance for response to a Fuel Handling Accident will be updated to recommend that the operators evaluate what dose minimization method for the Control Room is best suited for the case at hand (filtration or re-initiation of normal intake), then take the appropriate ventilation measures to minimize dose. [See Commitment 3].
4.2.5 Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	No credit was taken for the use of personal protective equipment or prophylactic drugs.
4.2.6 The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	The dose receptor for these analyses was the hypothetical maximum exposed individual, who is present in the Control Room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual was assumed to be 3.5×10^{-4} cubic meters per second.
4.2.7 Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose ..."	The Control Room doses were calculated using the same dose conversion factors as identified in Regulatory Position 4.1 for use in offsite dose analyses. Also, the RADTRAD computer code uses the equation provided in Section 4.2.7 for correcting the finite versus semi-infinite cloud assumptions.
4.3 Other Dose Consequences "The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source	See Section 1.3.1 above for the responses to each of those items. "Design envelope source terms" are not being changed by the Fuel Handling Accident dose re-calculation. Radiation exposure estimates to plant personnel for many of the NUREG–0737

Regulatory Guide 1.183 Guidance	Degree of Conformance
terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide."	considerations are also not affected by a Fuel Handling Accident. The Technical Support Center doses were addressed through a scoping study comparison to the Control Room. Equipment qualification requirements for plant equipment in the Fuel Handling Building are not being revised as a result of the new Fuel Handling Accident calculation, consistent with guidance in Regulatory Guide 1.183, Section 1.3.5. In the Containment, the Fuel Handling Accident doses are not bounding for EQ purposes, so the design basis integrated exposure values are unaffected.
4.4 Acceptance Criteria "The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6. The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii)."	The 5 rem TEDE Control Room dose criterion from 10 CFR 50.67 is used. For EAB and LPZ, the 6.3 rem TEDE dose criterion from Table 6 of Regulatory Guide 1.183 is used (~25% of the 10 CFR 50.67 criterion). The NUREG-0737 item potentially affected by a Fuel Handling Accident is TSC dose (if the TSC is activated for such an event), which is estimated by a scoping study to be well within the 5 rem TEDE dose. The USAR markup provided in Enclosure 5 shows how the new dose criteria are being updated. RG 1.183 Table 6 also shows an "analysis release duration" of 2 hours. Instead of the 2-hour release duration, the calculation conservatively used an instantaneous (puff) release assumption.
Section 5. Analysis Assumptions and Methodology ***5.1 General Considerations*** *5.1.1 Analysis Quality* "The evaluations required by 10 CFR 50.67 ... should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	The revised Fuel Handling Accident calculations were prepared under a 10 CFR 50 Appendix B quality assurance program.
These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large	The conservative, bounding characteristics of the AST that the NRC staff chose to present in Regulatory Guide 1.183 are used in the calculations.

Regulatory Guide 1.183 Guidance	Degree of Conformance
uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence -- the proposed deviation may not be conservative for other accident sequences."	Therefore there are no proposed deviations from the AST characteristics that are based on specific accident sequences that would require additional justification to prove they are conservative for other accident sequences.
5.1.2 Credit for Engineered Safeguard Features "Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences."	No credit for ESF systems or components is taken in the base calculation, which produced acceptable results after 24 hours of radiological decay. In the Control Room base calculation, the normal Control Room ventilation system is considered to continue to run throughout the event without filtration (see discussion in Section 5.1.3 below).
5.1.3 Assignment of Numeric Input Values "The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications. ..."	Conservative assumptions were utilized in the analyses. As described above, one area in which sensitivity studies were completed is with the Control Room dose. The base case assumes the normal ventilation system continues to run. This ensures the intake of activity into the Control Room is maximized, and ensures no credit is taken for active functions such as isolations from the radiation monitor or activation of the Emergency Recirculation system. This base case conservatively assumed intake flow 10% above nominal in order to maximize the amount of activity that enters the Control Room, then conservatively assumed exhaust flow 10% below nominal after the activity has been introduced into the Control Room. The sensitivity studies examined actions the operators could take after a period of time in an isolated, non-filtered mode (2 hours after all the activity is introduced into the Control Room in the studies), to ensure that their choice of action would not result in a dose outlier. The sensitivities studied the effects of turning on the filtration system, with low carbon filter efficiency (50%), or re-establishing the normal system intake, effectively purging the Control Room. Both cases provided doses within the dose limits.

Regulatory Guide 1.183 Guidance	Degree of Conformance
5.1.4 Applicability of Prior Licensing Basis "The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Two items may be considered to be "retained items" from the current licensing basis. The first is the allowance that the water level above the reactor vessel flange may be 22 feet 9 inches, less than the standard 23 foot value. This has been previously reviewed and approved by the NRC based on the fact that there is actually no fuel stored at the level of the flange (there is more than 51 feet of coverage over the top of the fuel that is down in the reactor vessel itself). Technical Specification 3.9.6 requires the 22 foot 9 inch height over the flange of the reactor vessel. As explained in the Bases, a dropped bundle would not be striking another fuel bundle at this level where less than 23 feet of coverage exists. This limits the potential damage from the strike at this elevation to the pins in just one bundle rather than the two or more bundles that are involved in the bounding calculation (where a strike occurs in the core with 51 feet of coverage). By itself, this single versus multiple bundle damage limits the release at this height, and more than compensates for the coverage being less than 23 feet. Also, a bundle dropped at this elevation is falling less than 2 feet, rather than the drop of 34 feet assumed in the evaluation that determines the number of fuel pins that might be damaged by a drop. As already explained in the Technical Specification 3.9.6 Bases, the reduction in this water level over the flange is acceptable. To validate this conclusion, a separate calculation was performed (and is included as Appendix A to Enclosure 4) for the drop of a bundle that strikes the refueling shield. The refueling shield, which sets on the reactor vessel flange during the refueling process, is the highest horizontal surface that a fuel bundle could strike if dropped in the reactor cavity area. As expected, the resultant doses were bounded by the analyses where a dropped bundle hit multiple other bundles (the doses from the drop onto the refueling shield would be less than 75% of the design basis cases). Therefore, despite the water level being less than 23 feet, this does not represent the limiting case. Further details on the drop onto the refueling shield are contained in Appendix A to Enclosure 4. The second "retained item" from the current licensing basis is that the "Decay Time" specification will remain in the PNPP Operational Requirements Manual (ORM) The Decay Time specification was relocated out of the Technical Specifications as part of Amendment 69, the improved Technical Specifications. The Decay Time specification

Regulatory Guide 1.183 Guidance	Degree of Conformance
	requires that the plant be subcritical for at least 24 hours before movement of irradiated fuel may begin. It is proposed that this control remain in the ORM, since the PNPP Decay Time specification was relocated to the ORM as part of the improved Technical Specifications. The NRC Safety Evaluation for Amendment 69 still holds true, where it stated: "Although Criterion 2 of the Final Policy Statement would require [the Decay Time specification] to be retained in the improved TS, the requirement for a 24 hour decay time following subcriticality before commencing movement of irradiated fuel in the reactor vessel will always be met for a refueling outage. ...Therefore, the requirement is unnecessary and has been relocated from the specifications to the ORM."
5.2 Accident-Specific Assumptions "The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. ... Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses, or, in some cases, a previously approved licensing basis consideration. The assumptions in the appendices are deemed consistent with the AST identified in Regulatory Position 3 and internally consistent with each other. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect this consistency. ..."	Reg. Guide 1.183 Appendix B is the applicable appendix for a Fuel Handling Accident. Each assumption in that guidance is addressed below. Except for the 23 feet of water over the vessel flange issue discussed above, and the instantaneous puff release, also discussed above, alternatives to the assumptions in Appendix B are not being proposed.
5.3 Meteorology Assumptions "Atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. .. All changes in X/Q analysis methodology should be reviewed by the NRC staff."	No changes to Chi/Q atmospheric dispersion values are being proposed. The current USAR Chi/Q values for the Control Room, EAB and LPZ were approved in conjunction with License Amendments 102 and 103, in March 1999. The actual values used are presented in the calculation attached as Enclosure 4.
Section 6. Assumptions for Evaluating the Radiation Doses for Equipment Qualification "The assumptions in Appendix I to this guide are	No changes are proposed to equipment

Regulatory Guide 1.183 Guidance	Degree of Conformance
acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue."	qualification requirements at PNPP as a result of the reanalysis of the Fuel Handling Accident. Since most of the particulate radionuclides that escape from the fuel rod gap are assumed to convert to an elemental form prior to release from the water (see the Appendix B discussions below), the source term composition is not significantly different than before, except it has been scrubbed more efficiently by the water in the pool (DF of 200 versus 100). This more efficient scrubbing reduces the overall release above the pools, which is the dose that might be seen by plant equipment. In the Fuel Handling Building, the dose received by equipment from a Fuel Handling Accident would therefore be lower than the dose using the original assumptions. Also, in both the Containment and the Fuel Handling Building, the LOCA is the event that sets the EQ requirements for equipment, rather than the Fuel Handling Accident. Finally, after 24 hours of radiological decay, no credit is taken in the revised calculations for the operation of any plant equipment to mitigate the release before it escapes. Therefore it is conservative to retain existing EQ program requirements.
Appendix B ***ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT*** ***App. B Section 1. Source Term*** "Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The following assumptions also apply. *1.1* The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	The number of fuel rods damaged in a Fuel Handling Accident (151) was determined by the fuel vendor, Global Nuclear Fuels (GNF). The vendor used a methodology that has been generically reviewed and approved by the NRC as part of NEDE-24011-P-A (GESTAR II). The analysis considers the weight of a dropped GE 12 or 14 fuel assembly, including the weight of the triangular fuel handling mast. It also considered the height of the drop, and the compression, torsion and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies was also considered.
1.2 The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	The total core inventory is multiplied by the gap fractions from Position 3.2, a radial peaking factor, and the fraction of fuel rods failed. There are approximately 64,208 fuel rods in the core. The percentage of rods breached is 151/64,208 32, which is 0.235% of the core. This activity is instantaneously released. The radionuclide groups listed in Reg. Guide 1.183 Table 3 were considered.

Regulatory Guide 1.183 Guidance	Degree of Conformance
1.3 The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool."	No attempt is made to justify a mechanistic treatment of the halogen release from the pool. The non-organic halogens are assumed to re-evolve in an elemental form.
App. B Section 2. Water Depth "If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1)."	The only case where the water is not 23 feet or greater is over the reactor vessel flange/refueling shield, as discussed above. The tops of all the irradiated fuel assemblies in storage have greater than 23 feet of coverage. Specifically, the top of the fuel in the reactor vessel is ~ 51 feet below the surface, and the fuel in the Fuel Handling Building is ~ 28 feet below the surface. The fuel in the upper Containment pools is ~ 27 feet below the surface. Therefore, in all the calculations except for the "less than 23-foot" refueling shield calculation discussed above, an overall effective decontamination factor (DF) of 200 is used for the halogens (iodines and bromines). It should be noted that this is a conservatism, since a DF of 500 for the elemental species would actually result in a higher overall effective DF than 200, as discussed in several other plant's submittals on this subject. For the refueling shield calculation, an overall DF was calculated to be 152.4. Despite the reduced DF, this was shown not to be a limiting case, primarily due to fewer pins being damaged in this event as compared to the bounding event over the reactor vessel. Details are provided in Appendix A to Enclosure 4.
App. B Section 3. Noble Gases (and particulates) "The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor)."	100% of the Noble Gases (xenons and kryptons) are assumed to escape the water pool (DF of 1). None of the particulate radionuclides (the alkali metals – cesiums and rubidiums) are assumed to escape the water pool (DF of ∞).
App. B Section 4. Fuel Handling Accidents Within The Fuel Building "For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff.	The following section (Section 5) addresses "Fuel Handling Accidents Within Containment". At PNPP, a drop within Containment bounds the drop within

Regulatory Guide 1.183 Guidance	Degree of Conformance
4.1 The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period. 4.2 A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses. 4.3 The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums."	the Fuel Handling Building. Therefore the next section, which addresses the Containment, will provide more details on the bounding calculation. The Containment drop is bounding because the drop over the reactor vessel would have higher kinetic energy and therefore a greater number of individual fuel rods damaged. The drop distance over the vessel is ~ 30 feet (GE used 34 feet in their calculation), whereas in the Fuel Handling Building, the drop is ~ 8 feet. Since both analyses then assume that the activity which escapes from the pool is treated the same, i.e., it is released immediately and directly to the environment, the FHA inside Containment will be bounding. There also is no practical difference between the two buildings at PNPP during handling of fuel that is not considered to be "recently irradiated". After License Amendment 102, handling of fuel that has been subcritical for more than 7 days has been performed using a two building "envelope" consisting of the Containment and the FHB. The equipment hatch between these two buildings is opened, and the Fuel Handling Building Ventilation System can draw down the two building envelope. This exhaust is then routed through filters and out of the plant vent (note again that the calculations do not credit this filtration or delay time in the release).
App. B Section 5. Fuel Handling Accidents Within Containment "For fuel handling accidents postulated to occur within the containment, the following assumptions are acceptable to the NRC staff. *5.1* If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	No credit is taken for Primary or Secondary Containment isolation during fuel handling after 24 hours of radiological decay.
5.2 If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on ..."	No credit is taken for automatic isolations of the Primary or Secondary Containment after 24 hours of radiological decay.
5.3 If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Rather than releasing the activity over a 2-hour time period, the release is conservatively considered to be instantaneous.
5.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection,	No credit is taken for filter systems after 24 hours of radiological decay, although such systems will continue to be available during fuel handling per controls implemented for Amendment 102.

Regulatory Guide 1.183 Guidance	Degree of Conformance
actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	
5.5 Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. ..."	No credit is taken for dilution or mixing of the activity inside the Containment.

Regulatory Analysis/Commitments

The NRC's traditional methods (prior to the AST) for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides and Standard Review Plan (SRP) chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the ASTs and with the Total Effective Dose Equivalent (TEDE) criteria provided in 10 CFR 50.67. Regulatory Guide 1.183 provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. This guidance supersedes corresponding radiological analysis assumptions provided in the older regulatory guides and SRP chapters when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67. One of the Regulatory Guides that is superceded for PNPP for the Fuel Handling Accident is Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors".

Due to the comprehensive nature of Regulatory Guide 1.183, the matrix (Table) given above was incorporated into this submittal to show how each section of the new guidance is being addressed.

Also, the NRC published a new SRP section to address AST. It is Standard Review Plan Section 15.0.1, Rev. 0, entitled "Radiological Consequence Analyses Using Alternative Source Terms". It provides guidance on which NRC branches will review various aspects of an AST license amendment request, but otherwise is consistent with the guidance found in Regulatory Guide 1.183. The plant-specific information provided above to support the license amendment request is believed to adequately address the guidance found in SRP 15.0.1.

Several Regulatory documents other than Regulatory Guide 1.183 are applicable to the proposed change. The following matrix addresses these.

Other Regulatory Documents	
GDC 61, "Fuel storage and handling and radioactivity control." The fuel storage and handling ...systems ... shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and	The fuel storage and handling systems, including water coverage over the fuel, are not affected by the proposed changes. These systems can still be periodically inspected and tested. Radiation protection shielding by the buildings is also unaffected. Appropriate containment, confinement, and filtering systems will remain in place under controls implemented for License Amendment 102. [Note: the guidance on Fuel Handling Accidents has not required that releases be contained or confined. They are typically assumed to be released through a ventilation system within no longer than a 2-hour period. The ventilation systems that

Other Regulatory Documents	
filtering systems, ..."	serve the Containment area (Containment and Drywell Purge System, Annulus Exhaust Gas Treatment System, and –with the Containment equipment hatch removed- the exhaust subsystem of the Fuel Handling Building Ventilation System) all contain filtration systems.]
GDC 63, "Monitoring fuel and waste storage." Appropriate systems shall be provided in fuel storage ... systems and associated handling areas (1) to detect ... excessive radiation levels and (2) to initiate appropriate safety actions.	The controls implemented for License Amendment 102 will continue to ensure this monitoring is provided and appropriate safety actions are taken.
GDC 64, "Monitoring radioactivity releases." Means shall be provided for monitoring the reactor containment atmosphere, ..., effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.	The controls implemented for License Amendment 102 will continue to ensure monitoring of radioactivity releases is provided.
10 CFR 20, 10 CFR 50 Appendix I, Technical Specification 5.5.4 "Radioactive Effluent Controls Program", and Technical Specification 5.6.3 "Radioactive Effluent Release Report", each require monitoring of releases and limitations on their magnitude.	See above discussions. Systems and controls remain in place to meet these requirements.
Regulatory Guide 1.13, "Spent fuel storage facility design basis", Revision 1.	PNPP design conforms to this guide with the exception of paragraph C.4. The inventory of radioactive materials available for leakage is based on the assumptions given in Regulatory Guide 1.183.
Regulatory Guide 1.25, "Assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility for boiling and pressurized water reactors", Revision 0.	No longer applicable to PNPP. See Regulatory Guide 1.183 for the fuel handling accident.
Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", Revision 0.	PNPP conforms to this guide for the fuel handling accident, with minor exceptions to ♦ Appendix B, Section 2 (23-foot coverage over the reactor vessel flange, as addressed above in the discussions for Section 5.1.4 and App. B Item 2), and ♦ Table 6, and Appendix B, Sections 4.1 and 5.3 (an instantaneous release rather than a 2-hour release, as discussed above for Sections 4.1.5 and 4.4, and for App. B Items 4.1 and 5.3). The original PNPP licensing basis for a Fuel Handling Accident utilized Regulatory Guide 1.25.

Additional Licensing/Regulatory information is provided in the USAR markups in Enclosure 5.

The following table identifies the actions that are considered to be regulatory commitments. Any other actions discussed in this document represent intended or planned actions, are described for the NRC's information, and are not regulatory commitments. Please notify the Manager - Regulatory

Affairs at the Perry Nuclear Power Plant of any questions regarding this document or any associated regulatory commitments.

Commitments

1. The maximum core wide radial peaking factor of 2.0 is being added into the list of reload analysis parameters that must be re-verified each cycle.

2. The burnup limit of 62 GWD/MTU on the average exposure of the peak rod, and the LHGR limit of 6.3 kw/ft peak rod average power for the higher burnup fuel (> 54 GWD/MTU), are both being added into the list of reload analysis parameters that must be re-verified each cycle.

3. Procedural guidance for response to a Fuel Handling Accident will be updated to recommend that the operators evaluate what dose minimization method for the Control Room is best suited for the case at hand (filtration or re-initiation of normal intake), then take the appropriate ventilation measures to minimize dose.

Environmental Consideration

A review has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

SIGNIFICANT HAZARDS CONSIDERATION

A License Amendment for the Perry Nuclear Power Plant (PNPP) is proposed to reflect use of the Alternative Source Term (AST) outlined in Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", for fuel handling accident dose calculations. The revised calculations are utilized to determine at what point in time the fuel is no longer considered to be "recently irradiated fuel", i.e., when the regulatory dose acceptance criteria can be met without credit for filtration systems and the Containment/Fuel Handling Buildings.

The standards used to arrive at a determination that a request for amendment does not involve a significant hazard are included in Commission regulation 10 CFR 50.92, which states that operation of the facility in accordance with the proposed changes would not:

1) involve a significant increase in the probability or consequences of an accident previously evaluated; or
2) create the possibility of a new or different kind of accident from any accident previously evaluated; or
3) involve a significant reduction in a margin of safety.

The proposed amendment has been reviewed with respect to these three factors, and it has been determined that the proposed change does not involve a significant hazard because:

1. **This proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The proposed amendment involves implementation of the Alternative Source Term for the Fuel Handling Accident at PNPP. There are no physical design modifications to the plant associated with the proposed amendment. The revised calculations do not impact the initiators of a Fuel Handling Accident in any way. They also do not impact the initiators for any other design basis events. Therefore, because design basis accident initiators are not being altered by adoption of the Alternative Source Term analyses, the probability of an accident previously evaluated is not affected.

   With respect to consequences, the only previously evaluated accident that could be affected is the Fuel Handling Accident. The Alternative Source Term is an input to calculations used to evaluate the consequences of an accident, and does not by itself affect the plant response, or the actual pathway of the radiation released from the fuel. It does however, better represent the physical characteristics of the release, so that appropriate mitigation techniques may be applied. For the Fuel Handling Accident, the AST analyses demonstrate acceptable doses, within regulatory limits, after 24 hours of radiological decay, without credit for Containment/Fuel Handling building integrity, filtration system operability, or Control Room automatic isolation. Therefore, the consequences of an accident previously evaluated are not significantly increased.

   Based on the above conclusions, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **This proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

   The proposed amendment does not involve a physical alteration of the plant (no new or different type of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed changes). Also, no changes are proposed to the methods governing plant/system operation during handling of recently irradiated fuel, so no

new initiators or precursors of a new or different kind of accident are created. New equipment or personnel failure modes that might initiate a new type of accident are not created as a result of the proposed amendment.

Thus, this amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **This proposed amendment does not involve a significant reduction in a margin of safety.**

The proposed amendment is associated with the implementation of a new licensing basis for PNPP Fuel Handling Accidents. Approval of the change from the original source term to a new source term taken from Regulatory Guide 1.183 is being requested. The results of the accident analyses, revised in support of the proposed license amendment, are subject to revised acceptance criteria. The analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183. Safety margins have been evaluated and analytical conservatism has been utilized to ensure that the analyses adequately bound the postulated limiting event scenario. The dose consequences of the limiting Fuel Handling Accident remains within the acceptance criteria presented in 10 CFR 50.67 "Accident Source Term", and Regulatory Guide 1.183.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the Control Room, are within corresponding regulatory limits. For the Fuel Handling Accident, Regulatory Guide 1.183 conservatively sets the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) limits below the 10 CFR 50.67 limit, and sets the Control Room limit consistent with 10 CFR 50.67.

Since the proposed amendment continues to ensure the doses at the EAB, LPZ and Control Room are within corresponding regulatory limits, the proposed license amendment does not involve a significant reduction in a margin of safety.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above considerations, it is concluded that a significant hazard would not be introduced as a result of this proposed amendment.

FirstEnergy Perry Nuclear Power Plant	CALCULATION	Page i
	PNPP No. 6077 Rev. 2/7/01	NEI-0341

INITIATING DOCUMENT	CALCULATION TYPE	CALCULATION NO.
Project 99-001-31	AE Analysis	3.2.15.14, Rev. 0

TITLE/SUBJECT: Fuel Handling Accident Using Alternative Source Term

CLASSIFICATION	CATEGORY	REFERENCED IN USAR VALIDATION DATABASE?	REFERENCED IN ATLAS?	OPEN ASSUMPTIONS?
☒ SAFETY-RELATED	☒ ACTIVE			
☐ AUGMENTED QUALITY	☐ HISTORICAL	☒ YES	☒ YES	☐ YES
☐ NON SAFETY RELATED	☐ STUDY	☐ NO	☐ NO	☒ NO

COMPUTER PROGRAM(S)

RADTRAD Mod 3.02, Microsoft® Word 2000, Microsoft® Excel 2000

REVISION RECORD

Rev.	Description of Change	Affected Pages	Preparer/Date	☐ Reviewer/Date ☒ Verifier/Date	Lead Engineer/Date
0	Initial Issue	All	Harry A. Wagner 1/3/02	[illegible] 1/4/2002	[illegible] 1/4/02

INITIATING DOCUMENT	CALCULATION TYPE	CALCULATION NO.
Project 99-001-31	AE Analysis	3.2.15.14, Rev. 0

TITLE/SUBJECT: Fuel Handling Accident Using Alternative Source Term

OBJECTIVE OR PURPOSE:

The purpose of this calculation is to determine radiological consequences of a design basis fuel handling accident (FHA) at Perry Nuclear Power Plant (PNPP), which occurs at 24 hours following reactor shutdown. Total effective dose equivalent (TEDE) at the control room, exclusion area boundary (EAB), and outer boundary of the low population zone (LPZ) are to be calculated using a source term derived from NUREG-1465, Reg. Guide 1.183, NEI 99-03, and the following conservative assumptions: [DIN # 1, 2, 3]

- No credit for containment/fuel handling building integrity.
- No credit for Annulus Exhaust Gas Treatment System (AEGTS) or Fuel Handling Area Exhaust Ventilation System (FHAEVS).
- No credit for filtration of Control Room Emergency Recirculation System.
- No credit for the isolation of the control room intake.

This calculation will replace the PNPP FHA dose analyses for the EAB, LPZ, and control room, CEI calculations 3.2.8 and 3.2.8.1, which were performed using Reg. Guide 1.25 and TID-14844 methodologies. [DIN # 4, 5, 6, 7]

SCOPE OF CALCULATION/REVISION

This calculation performs radiological dose analysis at the control room, EAB, and outer boundary of LPZ for a design basis fuel handling accident using an alternative source term. The scope is limited to calculating TEDE for a given number of fuel rods failed.

SUMMARY OF RESULTS/CONCLUSIONS:

Table 7 lists TEDE values calculated for the control room, EAB, and LPZ and compares these with regulatory limits. As shown in Table 7, the RADTRAD calculated TEDE values are well below the regulatory limits. Table 8 shows the TEDE values calculated for control room for the two sensitivity cases assuming the initiation of 1) control room fresh air intake and 2) control room recirculation filtering at two hours after a control room isolation assumed to occur after intake of all the activity. Both cases show TEDE values below the regulatory limit for the control room. Appendix A concluded that the dose consequences for an FHA occurring while transiting fuel over the Refueling Chute is bounded by the dose consequences for the design basis FHA.

IMPACT ON OUTPUT DOCUMENTS:

This calculation will be the basis to revise the USAR and Technical Specifications.

DOCUMENT INDEX

DIN No.	Document Number/Title	Revision, Edition, Date	Reference	Input	Output
1	NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"	February 1995	☒	☐	☐
2	Reg. Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"	July 2000	☐	☒	☐
3	NEI 99-03, "Control Room Habitability Assessment Guidance," Nuclear Energy Institute	June 2001	☐	☒	☐
4	CEI CALC. No. Calculation 3.2.8, "FHA Inside Containment"	Rev. 1	☐	☒	☐
5	CEI CALC. No. 3.2.8.1, "Control Room Habitability Following a Fuel Handling Accident"	Rev. 0, 12/30/1998	☐	☒	☐
6	Reg. Guide 1.25 (Safety Guide 25), "Assumptions Used for Evaluating Radiological Consequences of a Fuel Handling Accident in a Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"	March 23, 1972	☒	☐	☐
7	TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites"	1962	☒	☐	☐
8	NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation"	June 1997	☒	☐	☐
9	NUREG/CR-6604, Supplement 1, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation"	June 8, 1999	☒	☐	☐
10	Letter from D. R. Rogers to J. B. Balcken, "Fission Product Inventories for Perry High Energy Cycles" (Attachment 2 to DIN # 12)	January 29, 1996; Revised, March 14, 1996	☐	☒	☐

DIN No.	Document Number/Title	Revision, Edition, Date	Reference	Input	Output
11	Code of Federal Regulations:				
	10 CFR Part 50.67, "ACCIDENT SOURCE TERM"	01/24/2000	☐	☒	☐
	10 CFR Part 50, Appendix A, "GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS"	01/24/2000	☒	☐	☐
	10 CFR Part 50, Appendix K	01/24/2000	☐	☒	☐
12	DI-240, "Fuel Handling Accident Input Assumptions: Fuels Input"	Rev. 0, 11-8-01	☐	☒	☐
13	ORNL/TM-7175, "A User's Manual for the ORIGEN 2 Computer Code"	July 1980	☒	☐	☐
14	Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion"	2nd Printing, 1989	☐	☒	☐
15	CCC-652 Oak Ridge National Laboratory RSICC Computer Code Collection MACCS2	V.1.12 Code Package, 1997	☐	☒	☐
16	Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil"	1993	☐	☒	☐
17	PNPP Drawings: 015-026 411-0103 413-0101 413-0102 414-0102 414-0523 4549-18-1 4549-0194-1		☐	☒	☐
18	Calculation PSAT.0840IT.03, "Perry Plant TEDE Calculation"	Rev. 5	☐	☒	☐

DIN No.	Document Number/Title	Revision, Edition, Date	Reference	Input	Output
19	License Amendment 102	March 1999	☐	☒	☐
20	License Amendment 103	March 1999	☐	☒	☐
21	License Amendment 112	June 2000	☐	☒	☐
22	NEDC-32868P, "GE-14 Compliance with Amendment 22 of NEDC-24011-P-A "GESTAR-II""	Rev. 1, December 11, 2000	☐	☒	☐
23	Perry Technical Specifications 3.9.6		☐	☒	☐
24	Letter from L. R. Conner of Global Nuclear Fuel to P. J. Curran of PNPP, "Fuel Handling Accident – Bounding Fuel Rod Pressure for GE12 and GE14" (Attachment 1 to DIN # 12)	November 5, 2001	☐	☒	☐
25	Calculation CL-M26-01	Rev. 1	☐	☒	☐
26	P&ID 912-610	Rev. CC	☐	☒	☐
27	Periodic Test Instruction PTI-GEN-P0011	Rev.1	☐	☒	☐
28	SCIENTECH Interoffice Memo from H. A. Wagage to T. Bladen, "RADTRAD Code Verification and Validation"	3/14/01	☒	☐	☐
29	Perry letter PY-CEI/NRR-1510L		☒	☐	☐
30	G. Burley, "Evaluation of Fission Product Release and Transport for Fuel Handling Accident," Radiological Safety Branch, Division of Reactor licensing	October 5, 1971	☒	☐	☐

TABLE OF CONTENTS

SUBJECT	PAGE
COVERSHEET:	i
OBJECTIVE OR PURPOSE	ii
SCOPE OF CALCULATION	ii
SUMMARY OF RESULTS/CONCLUSIONS	ii
IMPACT ON OUTPUT DOCUMENTS	ii
DOCUMENT INDEX	iii
CALCULATION COMPUTATION	1
1 METHOD OF ANALYSIS	1
2 ACCEPTANCE CRITERIA	4
3 ASSUMPTIONS	4
4 DETAILED CALCULATIONS	10
4.1 Development of Input Files	11
4.1.1 Main Input File, pnpp_fha.psf	11
4.1.2 Auxiliary Input Files	12
4.1.2.1 Input File on Release Fraction and Timing, pnpp_fha.rft	12
4.1.2.2 Input File on Radionuclides Inventory and Decay Data, pnpp_fha.nif	13
4.1.2.3 Input File on Dose Conversion Factors, pnpp_fha.dcf	13
4.2 Running RADTRAD Code	13
4.3 Results of the RADTRAD Run for the Base Case	13
4.4 Sensitivity Analysis	16
4.4.1 Sensitivity Case 1: Effect of CR Isolation and Fresh Air Intake	16
4.4.2 Sensitivity Case 2: Effect of CR Isolation and Recirculation Filtering	16
5 COMPUTER INPUT AND OUTPUT	16

Appendix A. Fuel Handling Accident while Transiting over the Refueling Shield

Attachment 1. Main RADTRAD Input File for PNPP FHA, pnpp_fha.psf: Plant Model, Release Scenario, and Output Flags
Attachment 2. Auxiliary RADTRAD Input File for PNPP FHA, pnpp_fha.rft: Release Fraction and Timing
Attachment 3. Auxiliary RADTRAD Input File for PNPP FHA, pnpp_fha.nif: Radionuclides Inventory and Decay Data
Attachment 4. Auxiliary RADTRAD Input File for PNPP FHA, pnpp_fha.dcf: Dose Conversion Factors
Attachment 5. RADTRAD Output File for PNPP FHA, pnpp_fha.out
Attachment 6. RADTRAD Output File for Sensitivity Case 1: Effect of CR Isolation and Fresh Air Intake
Attachment 7. RADTRAD Output File for Sensitivity Case 2: Effect of CR Isolation and Recirculation Filtering

LIST OF TABLES

Table 1. Dose Conversion Factors [DIN # 14, 15, 16] 6
Table 2. Atmospheric dispersion values (χ/Q) used (s/m^3) 8
Table 3. Gap Fractions for Fuel Handling Accident [Table 3, DIN # 2] 9
Table 4. Selection of RADTRAD Input Parameters for the Three Compartments 11
Table 5. Calculation of FHA Source Term 14
Table 6. Radionuclides Decay Data Used for the Analysis [DIN # 15] 15
Table 7. Comparison of TEDE Calculated for Control Room, EAB, and LPZ with Regulatory Limits 15
Table 8. Comparison of Sensitivity Analysis Results with the Base Case for TEDE Calculated for Control Room 16

LIST OF FIGURES

Figure 1. PNPP FHA Release Model 2

CALCULATION COMPUTATION

1 METHOD OF ANALYSIS

The RADTRAD computer code was used to determine the total effective dose equivalent (TEDE) at the control room, exclusion area boundary (EAB), and outer boundary of the low population zone (LPZ), for the design-basis fuel-handling accident (FHA) inside containment at the Perry Nuclear Power Plant (PNPP) using NUREG-1465 and Reg. Guide 1.183 alternate source terms. [DIN # 8, 9, 1, 2] When comparing an FHA inside containment with an FHA in the Fuel Handling Building, the inside containment event would have higher kinetic energy and greater number of fuel pins damaged. Both analyses make the equivalent assumption that the activity, which escapes from the pool, is released immediately and directly to the environment. Therefore, the present analysis was performed for an FHA inside containment, which will be bounding. Appendix A analyzed and concluded that the dose consequences for an FHA occurring while transiting fuel over the Refueling Shield is bounded by the dose consequences for the design basis FHA.

Although the RADTRAD computer code consists of standardized source term data, the PNPP-specific, GE-calculated core isotope inventory at 24 hours after reactor shutdown was used instead, along with guidance provided in Reg. Guide 1.183. [DIN # 10, 2] All the isotopes of bromine, iodine, krypton, and xenon with core activity greater than 1E-9 Ci/MWt (a total of 20) and their daughters, i.e., an additional three isotopes of cesium and rubidium, were used for the analysis. Thus a total of 23 isotopes were used for TEDE analysis. The source terms of isotopes of cesium and rubidium were ignored as they were assumed to be retained completely by the pool. Sprays and natural deposition that may reduce the quantity of radioactive material were not credited. No filters or deposition of radioactive material in any pathway was modeled. The analysis does not credit the isolation of control room intake following a signal from the Rad. Monitor. The analysis also considers the effects of isolating the CR intake after activity is introduced into the control room (i.e., trapping the activity in the control room). The analysis considers the effects of trapping the activity for two hours followed by the cleanup by either the Control Room Emergency Recirculation System or by reopening the control room intake. Radioactive decay and in-growth of radionuclide daughters were also modeled in this analysis.

The RADTRAD model estimates doses in the control room and at EAB and LPZ. (Figure 1 schematically shows the PNPP FHA Release Model.) The model calculates the changes in radioactivity in the containment as a result of releasing radioactivity from the containment to the environment. Radioactive material is assumed to transport from the release point to the control room air intake, EAB, and LPZ without delay or deposition to the ground. Atmospheric dispersion of radioactivity during transport was accounted by using dispersion factors (χ/Q values). The change in radioactivity in the control room results from radioactivity entering the room with air intake, release of radioactivity with air exhaust, radioactive decay of nuclides in the control room. An additional mechanism of changing activity in the control room is filtering, which was not modeled for the base case calculation but was performed for sensitivity analysis (case 2) (§4.4.2).

Equation 1 shows the modeling of the change in radioactivity in the containment or control room, referred to as a "compartment" in the RADTRAD model. [§2.1.1, DIN # 8]

$$\frac{d}{dt}N_{n,i} = S_{n,i} + \sum_{\substack{j=1 \\ j\neq i}}^{K} F_{i,j}N_{n,j} + \sum_{\substack{j=1 \\ j\neq i}}^{K} \beta_{n,v}\lambda_v N_{v,i} - N_{n,i}\sum_{\substack{j=1 \\ j\neq i}}^{K} \left|F_{i,j}\right| - \lambda_n N_{n,i} - \eta_{n,i}F_{n,rec}N_{n,i} \qquad \text{Equation 1}$$

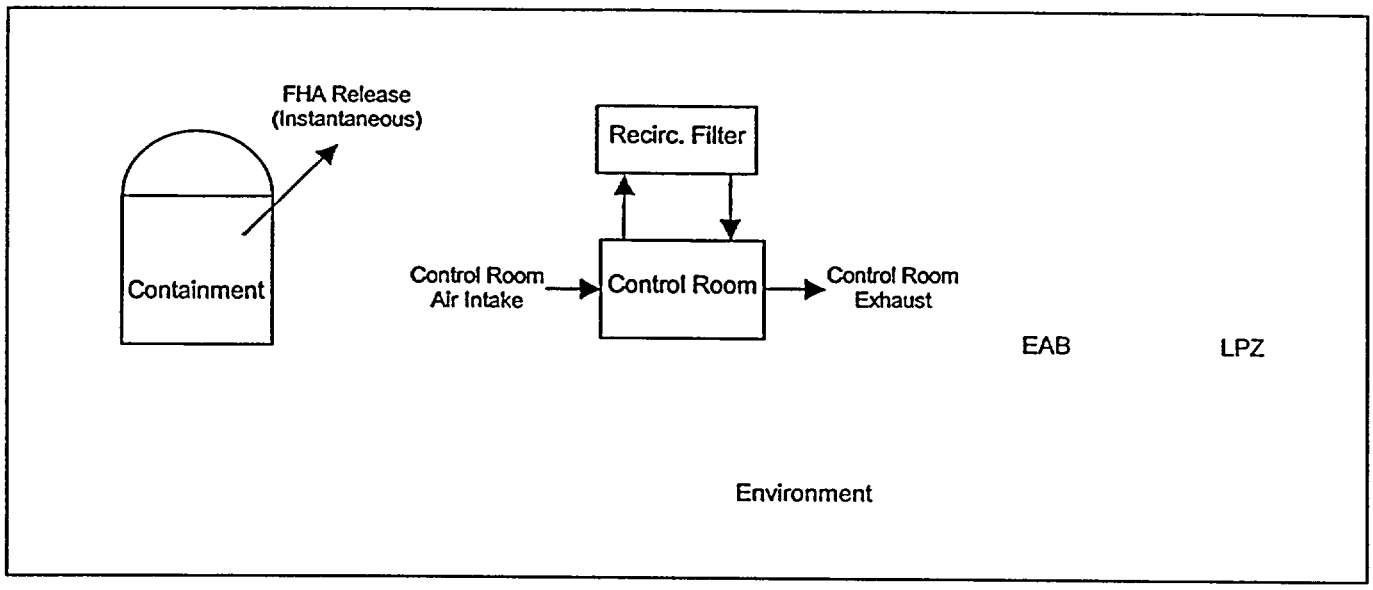


Figure 1. PNPP FHA Release Model[1]

where:

$N_{n,i}$ = number of atoms of nuclide n in compartment i

$S_{n,i}$ = source injection rate of nuclide n into compartment i (atoms/s)

K = Number of compartments defined in the RADTRAD model

$F_{i,j}$ = volume-normalized air flow rate from compartment j to i (s^{-1}) ($F_{i,j} \geq 0$)

$\beta_{n,v}$ = fraction of nuclide v that decays to nuclide n (dimensionless)

λ_n = radioactive decay constant of nuclide n (s^{-1}), which is calculated from radioactive half life, $(t_{1/2})_n$, (s) as shown in Equation 2

$\eta_{n,i}$ = filter efficiency for nuclide n in compartment i (dimensionless)

$F_{n,rec}$ = volume-normalized recirculation air flow rate in compartment i (s^{-1}).

$$\lambda_n = \frac{\ln(2)}{(t_{1/2})_n} \qquad \text{Equation 2}$$

The terms on the right hand side of Equation 1 models the following:

- Injection rate of radioactive source of radionuclide n into compartment i
- Intake rate of radionuclide n from all the other compartments into compartment i
- Generation rate of radionuclide n by decay of radionuclide v in compartment i
- Release rate of radionuclide n as a result of air exhaust from compartment i to all the other compartments. Note that the net airflow rate into a compartment is equal to the net air exhaust rate.

[1] Note that control room recirculation filtering was not modeled for the base case calculation but used only for sensitivity analysis (case 2) (§4.4.2)

- Decay rate of radionuclide *n* in compartment *i*
- Removal rate of radionuclide *n* from compartment *i* by recirculation filtering.

The TEDE was calculated as the sum of committed effective dose equivalent (CEDE) from inhalation and the effective dose equivalent (EDE), which is assumed to be equivalent to deep dose equivalent (DDE) from external exposure from each nuclide, as shown in Equation 3 (§3.13.4).

$$TEDE^{L} = \sum_{n=1}^{M}\left(EDE_n^L + CEDE_n^L\right) \qquad \text{Equation 3}$$

where *L* represents the location (control room, EAB, or LPZ) and *M* is the total number of radionuclides used in the analysis.

The EDE from each nuclide at environment (env) (EAB or LPZ) is calculated as given in Equation 4. [§2.3.1, DIN # 8]

$$EDE_n^{env} = DCF_{EDE,n}\int_0^T \dot{A}n\left(\chi/Q\right)_{env} dt \qquad \text{Equation 4}$$

where:

EDE_n^{env} = EDE (cloudshine dose) due to nuclide *n* in the environment at given location (rem)

$DCF_{EDE,n}$ = user-provided EDE (cloudshine) dose conversion factor for nuclide *n* $\left(\frac{\text{rem.m}^3}{\text{Ci.s}}\right)$.

T = duration of analysis (s)

$\dot{A}n$ = activity release rate of nuclide *n* (Ci)

$\left(\chi/Q\right)_{env}$ = user-provided atmospheric relative concentration at EAB or LPZ (s/m^3)

t = time (s)

The activity is related to the number of atoms of nuclide n as given in Equation 5.

$$A_n = N_n \lambda_n \qquad \text{Equation 5}$$

The CEDE from each nuclide at environment (env) (EAB or LPZ) is calculated as given in Equation 6. [§2.3.1, DIN # 8]

$$CEDE_n^{env} = DCF_{CEDE,n}\int_0^T \dot{A}n\left(\chi/Q\right)_{env} BR_{env}\, dt \qquad \text{Equation 6}$$

where:

$CEDE_n^{env}$ = CEDE due to nuclide *n* in the environment at given location (rem)

$DCF_{CEDE,n}$ = user-provided CEDE conversion factor for nuclide *n* (rem/Ci).

BR_{env} = user-provided breathing rate for the hypothetical individual at EAB or LPZ (m^3/s).

The EDE from each nuclide in the control room (CR) is calculated as given in Equation 7. [§2.3.2, DIN # 8]

$$EDE_n^{CR} = \frac{DCF_{EDE,n}}{G_F V_{CR}} \int_0^T A_{CR,n}\, OF\, dt \qquad \text{Equation 7}$$

where:

EDE_n^{CR} = EDE (cloudshine dose) due to nuclide *n* in the control room (rem)
G_F = geometry factor as calculated using Equation 8 (dimensionless)
$A_{CR,n}$ = activity of nuclide *n* in the control room at time *t* (Ci)
V_{CR} = Volume of the control room (m^3)
OF = user-provided control room occupancy factor (dimensionless)

$$G_F = \frac{1173}{V_{CR}^{0.338}} \qquad \text{Equation 8}$$

where:

V_{CR} - Volume of the control room (ft^3). *(Note the difference of units of this variable in Equation 7 and Equation 8.)*

The CEDE from each nuclide in the control room is calculated as given in Equation 9. [§2.3.2, DIN # 8]

$$CEDE_n^{CR} = \frac{BR_{CR}\, DCF_{CEDE,n}}{V_{CR}} \int_0^T A_{CR,n}\, OF\, dt \qquad \text{Equation 9}$$

where:

$CEDE_n^{CR}$ = CEDE due to nuclide *n* in the control room (rem)
BR_{CR} = user-provided breathing rate for the control room operator (m^3/s).

2 ACCEPTANCE CRITERIA

Both EAB and LPZ dose limits for FHA are TEDE of 6.3 rem during the analysis release duration of 2 hours. [Table 6, DIN # 2] The control room dose limit for FHA is TEDE of 5 rem for the duration of the accident (10CFR50.67(b)(2)(iii)). [DIN # 11]

3 ASSUMPTIONS

3.1 The FHA was assumed to occur at 24 hours following reactor shutdown.
3.2 When comparing an FHA inside containment with an FHA in the Fuel Handling Building, the inside containment event would have higher kinetic energy and greater number of fuel pins damaged due to the comparative height of the drop. Both analyses make the equivalent assumption that the activity, which escapes from the pool, is released immediately and directly to the environment. Therefore, the present analysis was performed for an FHA inside containment, which will be bounding.
3.3 No integrity of containment/fuel handling building was assumed.

3.4 No credit was taken for Annulus Exhaust Gas Treatment System (AEGTS) or Fuel Handling Area Exhaust Ventilation System (FHAEVS).

3.5 No credit was taken for filtration of Control Room Emergency Recirculation System during the base case calculation. The effect of control room recirculation filtering following isolation of the control room air intake was studied in sensitivity case 2 (§4.4.2).

3.6 No control room isolation was assumed for the base case calculation. Sensitivity cases were run with RADTRAD to assess the impact of isolating the control room after the activity has entered the control room (§4.4).

3.7 All failed fuel is assumed to be operating at high peaking factors and maximum exposures although the core operating limits on power density would prohibit high-exposure bundles from being at high peaking factors.

3.8 This analysis assumed a radial peaking factor of 2. [DIN # 12]

3.9 Radionuclide release from FHA was assumed to occur instantaneously.

3.10 Radioactive decay and corresponding in-growth of radionuclide daughters were modeled during this analysis.

3.11 Radioactive material is assumed to transport from the release point to the control room air intake, EAB, and LPZ without delay or deposition to the ground.

3.12 Fission Product Inventory (§3.1, Reg. Guide 1.183): [DIN # 2]
Reg. Guide 1.183 states that the inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed rated thermal power times the ECCS evaluation uncertainty.[2] The period of irradiation should be sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2. [DIN # 13]
This analysis used the ECCS evaluation uncertainty of 1.02. Fission product inventories were calculated by GE using the ORIGEN 2 computer code assuming 1500 effective full power days (EFPD) of operation. [DIN # 10]

3.13 Offsite Dose Consequences (§4.1, Reg. Guide 1.183): [DIN # 2]

3.13.1 This calculation determines TEDE, which is the sum of CEDE from inhalation and EDE, which is assumed to be equivalent to DDE from external exposure (Equation 3) (§3.13.4). Impact of daughter products was considered by decaying core radionuclides inventory for 24 hours in DIN # 10. Radioactive decay and corresponding in-growth of radionuclide daughters were modeled during this analysis (§3.10).

3.13.2 This calculation applies the CEDE conversion factors from Federal Guidance Report 11, which are readily available in the dose conversion factors file for 825 radionuclides, Dosdat825.inp, which was provided with the MACCS2 computer code package. [DIN # 14, 15] The CEDE conversion factors for the 23 nuclides that were selected for TEDE analysis, as described in §4.1.2.2, are listed in Table 1. Note that for BR 82 and BR 83, this analysis used more conservative CEDE conversion factors, which are for the lung clearance class of W (weeks) as given in DIN # 14 than those used in DIN # 15 for the class D (days).

3.13.3 This calculation applies the recommended breathing rates: 3.5E-4 m^3/s for the first 8 hours, 1.8E-4 m^3/s from 8 to 24 hours, and 2.3E-4 m^3/s thereafter. However, since the release was conservatively modeled as an instantaneous release, the analysis used an effective breathing rate of 3.5E-4 m^3/s.

3.13.4 The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective

[2] The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02. [DIN # 11]

dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining dose to the TEDE.

This calculation applies the EDE conversion factors from Federal Guidance Report 12, which are readily available in the dose conversion factors file for 825 radionuclides, Dosdat825.inp, which was provided with the MACCS2 computer code package. [DIN # 16, 15] The EDE conversion factors for the 23 nuclides that were selected for TEDE analysis, as described in §4.1.2.2, are listed in Table 1.

Table 1. Dose Conversion Factors [DIN # 14, 15, 16]

No.	Isotope	EDE		CEDE	
		(rem-m^3/Ci-s)	(Sv-m^3/Bq-s)	(rem/Ci)	(Sv/Bq)
1	BR 82	4.810E-01	1.300E-13	1.528E+03	4.130E-10
2	BR 83	1.413E-03	3.820E-16	8.917E+01	2.410E-11
3	KR 83M	5.550E-06	1.500E-18	0	0
4	KR 85	4.403E-04	1.190E-16	0	0
5	KR 85M	2.768E-02	7.480E-15	0	0
6	KR 87	1.524E-01	4.120E-14	0	0
7	KR 88	3.774E-01	1.020E-13	0	0
8	RB 87	6.734E-06	1.820E-18	3.234E+03	8.740E-10
9	RB 88	1.243E-01	3.360E-14	8.362E+01	2.260E-11
10	I129	1.406E-03	3.800E-16	1.735E+05	4.690E-08
11	I130	3.848E-01	1.040E-13	2.642E+03	7.140E-10
12	I131	6.734E-02	1.820E-14	3.289E+04	8.890E-09
13	I132	4.144E-01	1.120E-13	3.811E+02	1.030E-10
14	I133	1.088E-01	2.940E-14	5.846E+03	1.580E-09
15	I134	4.810E-01	1.300E-13	1.314E+02	3.550E-11
16	I135	3.069E-01	8.294E-14	1.228E+03	3.320E-10
17	XE129M	3.922E-03	1.060E-15	0	0
18	XE131M	1.439E-03	3.890E-16	0	0
19	XE133	5.772E-03	1.560E-15	0	0
20	XE133M	5.069E-03	1.370E-15	0	0
21	XE135	4.403E-02	1.190E-14	0	0
22	XE135M	7.548E-02	2.040E-14	0	0
23	CS135	2.091E-06	5.650E-19	4.551E+03	1.230E-09

3.13.5 For the EAB, the objective of the TEDE analysis is to consider the dose during the worst two hour period. The maximum allowed duration of release from an FHA is 2 hours. [Table 6, DIN # 2] For this analysis, the release was assumed to transport instantaneously from the release location to the receptor. Because the complete release was assumed to be instantaneous, the initial 2-hour gives the maximum dose. Therefore, TEDE was determined for the first two hours and no sliding window calculations were performed.

3.13.6 The CFR states that the TEDE should be determined for the most limiting receptor at the outer boundary of the LPZ and should be used in determining compliance with the dose criteria in 10 CFR 50.67. [DIN #11]
This calculation determined the TEDE for the first two hours for the most limiting receptor at the outer boundary of LPZ. The radioactivity was available at LPZ during the release, which occurred during the initial 1E-4 hours (0.36 s) because no delay was assumed for the transport of activity. Therefore, no TEDE was received by the receptor at LPZ after 0.36 seconds, as no activity was available.

3.13.7 The dispersion factors used in this calculation do not take credit for ground or any other deposition.

3.14 Control Room Dose Consequences (§4.2, Reg. Guide 1.183): [DIN # 2]

3.14.1 This calculation considers potential radiation sources to the control room operator:

3.14.1.1 Unfiltered intake of the radiation plume into the control room was assumed to occur at 6600 cfm (normal flow rate + 10%) during the release, which occurred during the initial 1E-4 hours (0.36 s). Exhaust flow rate was chosen to be equal to the intake flow rate. After radioactivity was taken into the control room, the exhaust flow rate was conservatively chosen to be at 5400 cfm (normal flow rate – 10%) in order to minimize the purging effect of the ventilation system. During this time, the intake flow rate was chosen to be equal to the exhaust flow rate.

3.14.1.2 Intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope:
Other than the 6600 cfm of unfiltered inleakage being assumed to be introduced directly into the control room from outside, infiltration of airborne radioactive material from adjacent areas and structures was considered to be a negligible dose contributor and was neglected. Normal operation maintains a positive differential pressure between the inside and outside of the control room and thus between adjacent spaces. Control room doors lead to closed chase spaces, closed stairwells, or closed corridor spaces such that neither outside wind conditions nor other ventilation systems can cause infiltration/leakage into the control room.

3.14.1.3 Radiation shine from the external radioactive plume released from the facility:
Radiation shine from the external radioactive plume for the purpose of this calculation was considered to be a negligible dose contributor. The roof of the control complex building consists of a 2'-4.5"-thick concrete slab and the control room ceiling is an 18"-thick concrete slab (PNPP Drawings 015-026 and 414-0523). [DIN # 17] Considering this shielding, the contribution to the control room dose due to the cloud passing by was considered to be negligible and was neglected. This judgment is further supported by the LOCA control room dose calculation, which documents the cloud direct gamma dose for 30 days as being <0.05% of the total control room LOCA dose. [DIN # 18] The percentage should be lower for a fuel handling accident and its resultant brief plume.

3.14.1.4 Radiation shine from radioactive material in the reactor containment:
The direct line from the containment with the least shielding is through the 3'-thick concrete containment shield building, the 2'-thick concrete control building wall and the 1.5'-thick concrete control room ceiling (PNPP Drawings 015-026, 414-0102, 411-0103). [DIN # 17] Other direct lines from the containment to the control room would result in additional concrete shielding. Similarly, the direct line from the fuel handling is through a 3'-thick concrete fuel handling area wall, a 3'-thick concrete Intermediate Building wall and a 2'-thick Control Complex building wall (PNPP Drawings 413-0101, 413-0102, 414-0102). [DIN # 17] Considering this shielding, the contribution to the control room dose due to shine resulting from radioactive material in the containment or fuel-handling building was considered to be negligible and was neglected. This judgment is further supported by the LOCA control room dose calculation which documents the containment direct gamma dose for 30 days as being ~3% of the total control room LOCA dose. [DIN # 18]

3.14.1.5 There are no additional sources of radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters. Radioactive material buildup in recirculation filters was not considered as the recirculation filters were not assumed to operate during FHA.
The effect of control room recirculation filtering following isolation of the control room air intake was studied in sensitivity case 2 (§4.4.2). The recirculation filters are located outside the control room envelope. The filter plenum equipment pad as well as the 18"-thick concrete control room ceiling shields the control room envelope. Therefore, radioactivity buildup in control room isolation filters would not affect the results of sensitivity case 2.

3.14.2 The radioactive material releases and radiation levels used in the control room dose analysis were determined using the same source term, transport, and release assumptions used for determining the EAB and LPZ TEDE values.

3.14.3 RADTRAD computer code was used to model transport of radioactive material into and through the control room. This modeling provides suitable conservative estimates of the exposure to control room personnel.

3.14.4 No credit was taken for engineered safety features that mitigate airborne radioactive material within the control room. The effect of trapping the activity in the control room for two hours, followed by cleanup by the emergency filters was studied in sensitivity case 2 (§4.4.2).

3.14.5 No credit was taken for using protective equipment or prophylactic drugs.

3.14.6 The dose receptor for these analyses was the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of this event, the breathing rate of this individual was assumed to be 3.5E-4 m^3/s.

3.14.7 Control room doses were calculated using the same dose conversion factors as the offsite dose calculation given in Table 1. RADTRAD computer code uses Equation 8 to correct the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors for the EDE from photons.

3.15 Acceptance Criteria (§4.4, Reg. Guide 1.183): [DIN # 2]
As given in §2, this calculation applies acceptance criteria in Table 6 of Reg. Guide 1.183 and 10CFR50.67 for the offsite and control room doses. [DIN # 2, 11] Instead of the 2-hour release duration that is recommended for FHA in Table 6 of Reg. Guide 1.183, this calculation conservatively used instantaneous release assumption.

3.16 Meteorology Assumptions (§5.3, Reg. Guide 1.183): [DIN # 2]
The analysis uses atmospheric dispersion values (χ/Q) used for the EAB, LPZ, and control room, listed in Table 2, that were previously approved by the NRC. [DIN # 19, 20]

Table 2. Atmospheric dispersion values (χ/Q) used (s/m^3)

Location	χ/Q (s/m^3)	Reference
Control Room	3.5E-4[a]	DIN # 5
EAB	4.3E-4	DIN # 4
LPZ	4.8E-5	DIN # 4

[a] DIN # 5 lists χ/Q values for different time periods up to 30 days. The initial value, which was for 0 to 8 hours, was used in this calculation because FHA release, and thus the activity intake to the control room was assumed to be instantaneous.

3.17 Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident (Appendix B, Reg. Guide 1.183): [DIN # 2]

3.17.1 Source Term:

3.17.1.1 The number of fuel rods damaged during an FHA, i.e., 151, was based on the fuel vendor's NRC approved methodology for GE 12 and 14 bundles and triangular fuel handling mast. [DIN # 22]

3.17.1.2 This calculation used the gap fractions in §3.2 of Reg. Guide 1.183, as listed in Table 3, and assumed that the source terms were instantaneously released. Only bromine, iodine, krypton, and xenon radioisotopes were used in the calculation of FHA source term. Cesium and rubidium radioisotopes were assumed to retain completely by the fuel pool water. Reg. Guide 1.183 noted that these gap fractions were applicable up to a peak rod average exposure of 62 GWD/MTU provided that the maximum linear heat generation rate did not exceed 6.3-kW/ft peak rod average power for burnup exceeding 54 GWD/MTU. As noted in DIN # 12, PNPP fuel designs satisfy this criterion.

Table 3. Gap Fractions for Fuel Handling Accident [Table 3, DIN # 2]

Isotope/Group	Gap Fraction
I-131	8%
Kr-85	10%
Other Noble Gases (Xe, Kr) and Halogens (I, Br)	5%
Alkali Metals[a] (Cs, Rb)	12%

[a] Alkali metals were assumed to be retained completely by the pool (§3.17.1.2, 3.17.3).

3.17.1.3 The chemical form of radioiodine released from the fuel to the spent fuel pool was assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The CsI release from the fuel was assumed to completely dissociate in the pool water. Because of the low pH of pool water, the iodine is assumed to re-evolve as elemental iodine. This was assumed to occur instantaneously.

As a halogen, bromine isotopes were modeled identical to iodine in terms of chemical form.

3.17.2 Water Depth:

This calculation used the Reg. Guide 1.183 pool overall DF value of 200 for iodine isotopes. (As a halogen, bromine isotopes were modeled identical to iodine in terms of pool DF.) This assumption requires that PNPP pools maintain at least 23 feet of water coverage above damaged fuel.

The PNPP requires that it maintain at least 23' of water coverage above the fuel in the reactor pressure vessel (RPV) or the spent fuel storage pool. The PNPP has approximately 51.5' of water above the core, 27' above the fuel rack in the upper containment pool, and ~ 28' of coverage over spent fuel in the spent fuel pool (25' above IFTS gate sill). Therefore, if the dropped fuel bundle strikes another irradiated fuel bundle, 23' of water coverage above the damage bundle will be available.

Per Technical Specifications, PNPP requires that only 22'-9" of water coverage above the RPV flange during refueling. If the dropped bundle were to strike the RPV flange versus another bundle, there would be a possibility that only 22'-9" of water coverage is available. [DIN # 23] However, as addressed in the bases for Technical Specifications 3.9.6, such a drop will result in reduced release of fission gasses and it was judged that slight reduction in water level was acceptable. In addition, the drop onto the RPV flange will be on the order of 1.5' to 2', which is significantly less than the drop of 34' that was assumed in the GE analysis, which calculated the number of fuel rods failing during an

FHA. Therefore, the actual number of fuel rods failing in such an event will be significantly less than that assumed for this analysis (i.e., 151 fuel rods) and thus, the slightly reduced water level would be acceptable.

3.17.3 Noble Gases/Particulates:
The retention of noble gases in the water in the fuel pool or reactor cavity was assumed to be negligible (i.e., decontamination factor of 1). Particulate radionuclides (Cs and Rb) were assumed to be retained by the water in the fuel pool or reactor cavity (i.e., decontamination factor of ∞).

3.17.4 Fuel handling Accidents within Containment:

3.17.4.1 It was conservatively assumed that the containment was not isolated during fuel handling operations.

3.17.4.2 It was conservatively assumed that the containment would not isolate in the event of an FHA.

3.17.4.3 For an open containment, Reg. Guide 1.183 recommends assuming that the radioactive material that escapes from the fuel building to be released to the environment over a 2-hour time period. This analysis conservatively assumed that the release took place instantaneously.

3.17.4.4 No credit was assumed for a reduction in the amount of radioactive material released from the containment by engineered safety features filter systems.

3.17.4.5 No credit was assumed for dilution or mixing of the radioactivity released from the reactor cavity by natural or forced convection inside the containment.

3.18 NEI 99-03 Insight on Release Pressure Limit of 1200 psig: [DIN # 3]
For pool overall DF value of 200 for iodine (and bromine) isotopes to be applicable, in addition to a 23' depth in the pool, the release pressure is to be limited to 1200 psig. (See §3.17.2.) In the event of an FHA, the PNPP fuel rod pressure will be below 1200 psig. [DIN # 24]

3.19 General Design Criteria for Nuclear Power Plants (10 CFR 50, Appendix A): [DIN # 11]
General Design Criteria (GDC) 61 and 63 of 10 CFR 50, Appendix A addresses FHA.

3.19.1 GDC 61-Fuel storage and handling and radioactivity control states that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed with appropriate containment, confinement, and filtering systems. At PNPP, these systems include the primary and secondary containment, Fuel Handling Building, and AEGTS/FHAEVS. However, this analysis assumes no credit for the presence of these buildings or AEGTS/FHAEVS.

3.19.2 GDC 63-Monitoring fuel and waste storage states that appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions. This analysis assumes no credit for the initiation of appropriate safety actions.

4 DETAILED CALCULATIONS

Calculations were performed using the RADTRAD computer code, which requires a main input file and three auxiliary input files. The main input file, named as pnpp_fha.psf in this analysis, describes 1) the plant model with compartments and release pathways, 2) the release scenario with source term, release rates, mechanisms of reducing radioactive elements, including overlying pools, suppression pool, sprays, filters, and natural deposition, and 3) the code output. The three auxiliary input files, named as pnpp_fha.rft, pnpp_fha.nif, and pnpp_fha.dcf in this analysis, describe release fraction and timing, radionuclides inventory, and radioactivity to dose conversion factors. Section 4.1 describes the development of input files. Sections 4.2 and 4.3 describe running and results of the RADTRAD code

4.1 Development of Input Files

4.1.1 Main Input File, pnpp_fha.psf

The input parameters are described in this section in the same order as they appear in the main input file, pnpp_fha.psf, which is given in Attachment 1. The radionuclides inventory file was defined in the main input file as pnpp_fha.nif, which is described in §4.1.2.2. Power level was set to 3833.2 MWt (102% of 3758 MWt). [DIN # 21]

Figure 1 schematically shows the PNPP FHA Release Model, which consists of three compartments, the containment, control room, and environment, and three-release/flow paths, release from the containment to the environment, control room air intake, and control room exhaust.

Table 4 shows the selection of RADTRAD input parameters for the three compartments. Each value of the compartment type is that recommended in the RADTRAD input manual. [DIN # 9] Volume of the containment was arbitrarily selected as 1 ft^3 because the actual value is unimportant for the assumed instantaneous, complete release of the FHA source term. Following example problems given in the RADTRAD input manual, the volume of the environment was chosen as zero. [DIN # 8]

Table 4. Selection of RADTRAD Input Parameters for the Three Compartments

Number	Name	Type [DIN # 9]	Volume (ft^3)
1	Containment	3	1.0 (arbitrary)
2	Environment	2	0
3	Control Room	1	367,070 [DIN # 5, 25]

The three radioactivity release/intake pathways are identified between the compartments as schematically shown in Figure 1. The type of release pathway from the containment to the environment was defined as "air leakage" by choosing number 4 for the release flag. [DIN # 9]

The type of release pathway for 1) air intake from the environment to the control room and 2) air exhaust from the control room to the environment was defined as "filtered pathway" by choosing number 2 for the release flag. [DIN # 9] Note that the filter efficiencies were set to zero for both of these two paths, later in the input file.

The number 1 below the line with "Source Term" identifies that the whole source term was placed in only one compartment. Numbers 1 and 1 in the next line identify that the whole source term was placed in the containment (compartment # 1).

The auxiliary input files giving dose conversion factors, and release fraction and timing are identified as pnpp_fha.dcf and pnpp_fha.rft. These files are described in §4.1.2.3 and 4.1.2.1.

Delay time for the release was chosen as zero in order to model the release as instantaneous. A value of 1 was chosen for the flag to enable the calculation of radioactive daughter products. Next line shows the fractions of aerosol, elemental, and organic halogens and the fraction of halogens that are radioactive as 0, 0.9985, 0.0015, and 1 (§3.17.1.3). (Note that the fraction of halogens that are radioactive is a redundant input required by RADTRAD because specifying the activity of a nuclide implies that it is radioactive.)

No overlying pools were modeled with RADTRAD because decontamination factors were modeled separately in calculating the FHA source term as described in §4.1.2.2.

The number of compartments is given as 3. The first and second numbers under each compartment give flags to indicate whether detail output was to be given and whether radioactive decay is to be calculated. Detailed output was requested only for the control room by choosing a flag value of 1. For all the three compartments, the containment, environment, and control room, radioactive decay was modeled by choosing a flag value of 1.

The dose calculation model was run for 30 days (720 hours). The release rate from the containment was chosen arbitrarily as 1E10 %-volume/day. A high value was chosen to ensure complete, instantaneous release (§3.9).

Both control room intake and exhaust flow rates were set to 6,600 cfm (6000 cfm + 10% is based on the design value as shown on DIN # 26 with an allowable operating tolerance as specified in DIN # 27) during activity intake, which occurred during the initial 1E-4 hours (0.36 s). After the activity was taken into the control room, both intake and exhaust flow rates were set to 5,400 cfm (6000 cfm - 10%). The filter efficiencies for both of intake and exhaust flow paths were set to zero.

TEDE was calculated at three locations, EAB, outer boundary of LPZ, and control room. Atmospheric dispersion values (χ/Q) of 4.3E-4 and 4.8E-5 s/m^3 were used for EAB and the outer boundary of LPZ (Table 2). For both offsite locations the Reg. Guide 1.183 recommended breathing rates were used: 3.5E-4 m^3/s for the first 8 hours, 1.8E-4 m^3/s from 8 to 24 hours, and 2.3E-4 m^3/s thereafter (§3.13.3). However, considering that the release was conservatively modeled as near an instantaneous release, the effective breathing rate used was the highest of the three values recommended, i.e., 3.5E-4 m^3/s.

Atmospheric dispersion value (χ/Q) of 3.5E-4 s/m^3 was used for the control room intake for 30 days (Table 2). The control room occupancy factors used were 100% during the first 24 hours after the event, 60% between 1 and 4 days, and 40% from 4 to 30 days (§3.14.6). A constant breathing rate of 3.5E-4 m^3/s was used for the control room (§3.14.6).

Simulation time steps were selected as follows: 0.025 h from 0 to 8 h, 0.1 h from 8 h to 24 h, and 0.4 h from 24 h to 720 h (30 days).

Flag value of 1 was selected for each to include plant model, scenario description, and results for every simulation in the output.

4.1.2 Auxiliary Input Files

4.1.2.1 Input File on Release Fraction and Timing, pnpp_fha.rft

The auxiliary input file on release fraction and timing, pnpp_fha.rft, given in Attachment 2, shows that 100% of noble gas (Kr and Xe), halogen (Br and I), and alkali metals (Cs and Rb) groups were released to the containment[3] in 1E-4 hours (0.36 s). The source term of Cs and Rb is zero because they were assumed to be retained completely by the pool, thus the timing and fraction of release for alkali metals group is immaterial. (Note that the input file identifies groups, except the noble gases group, by the representative nuclide in each group. Thus, the halogen group is named as iodine group.)

[3] During an FHA, activity in the gap is released into the fuel pool. The radionuclides that are not retained in the pool are immediately released from the top of the pool, in this case to the containment. The FHA source term, as listed in the last column of Table 5, was calculated by accounting for the pool DF. Therefore, the fuel pool was not specifically input to the RADTRAD model, and the complete source term was released to the containment but not to the pool.

4.1.2.2 Input File on Radionuclides Inventory and Decay Data, pnpp_fha.nif

The FHA source term was determined using GE-calculated core inventory for 641 radioisotopes for different decay times for 1500 EFPD of operation. [DIN # 10] These included 17 bromine, 21 iodine, 15 krypton, and 18 xenon isotopes, amounting to a total of 71 isotopes. Note that isotopes of alkali metals (cesium and rubidium) were ignored as they were assumed to be retained completely in the pool. Of 71 isotopes, 48 isotopes with zero activity at 24 hours after reactor shutdown were ignored. In addition three isotopes, BR 84, I128, and XE138, which had activity less than 1E-9 Ci/MWt at 24 hours after reactor shutdown, were also ignored. The remaining 20 isotopes of bromine, iodine, krypton, and xenon were chosen for TEDE analysis. Three additional nuclides, RB 87, RB 88, and CS135, which are daughter products of radioactive decay of KR 87, KR 88, and XE135M that were included in original set of 20 isotopes, were also used for TEDE analysis. Thus a total of 23 radionuclides of bromine, cesium, iodine, krypton, rubidium, and xenon were used for TEDE analysis. Table 5 lists the 23 isotopes used for this analysis and their core inventory at 24 hours after reactor shutdown. [DIN # 10]

Table 5 shows the calculation of FHA source term. The core inventory per unit power was multiplied by the gap fraction, reactor power, the fraction of fuel rods failed, and radial peaking factor and divided by the pool DF. The reactor power, fraction of fuel rods failed, and radial peaking factor were combined to form a single multiplication factor because each parameter was independent of isotopes. This calculation used a reactor power of 3833.2 MWt (102% of 3758 MWt) and a radial peaking factor of 2 (DIN # 21; §3.8). Of effective 64,208.32 rods in the core, this calculation assumed that 151 rods had failed during FHA (DIN # 12; §3.17.1.1). Using these values, the multiplication factor was calculated as 18.029 (= (3833.2 MWt)*((151 rods failed)/(64,208.32 rods in the core))*(2)).

Radionuclides decay data needed for this input file includes, identification of daughter nuclides and fractions and radioactive half-life as listed in Table 6. These data were obtained from the radionuclides data file for 825 radionuclides, Indexr.inp, which was provided with the MACCS2 computer code package. [DIN # 15]

4.1.2.3 Input File on Dose Conversion Factors, pnpp_fha.dcf

The input file on dose conversion factors, pnpp_fha.dcf, which is given in Attachment 4, was developed using the values listed in Table 1. Note that the units of EDE (cloudshine) and CEDE (inhaled chronic) dose conversion factors to be used for this file are in Sv-m^3/Bq-s and Sv/Bq.

4.2 Running RADTRAD Code

The RADTRAD computer code was installed and executed on a Dell Latitude computer running on Windows NT Version 4.0 operating system as currently assigned to Hanry Wagage (owned by Matrix Leasing, no. 210158). Satisfactory operation of the RADTRAD code on this computer has been confirmed by verification. [DIN # 28] The main input file, pnpp_fha.psf and the three auxiliary input files, pnpp_fha.rft, pnpp_fha.nif, and pnpp_fha.dcf were used as input to the code. These files are given in Attachment 1 through Attachment 4. The Output file, pnpp_fha.out, is given in Attachment 5.

4.3 Results of the RADTRAD Run for the Base Case

The detailed results are given in the computer output file, pnpp_fha.out, which is given in Attachment 5. Table 7 lists TEDE values calculated for the control room, EAB, and outer boundary of the LPZ and compare these with regulatory limits. As shown in Table 7, the RADTRAD calculated TEDE values are well below the regulatory limits.

Table 5. Calculation of FHA Source Term

No.	Isotope	Core Activity at 24 hours (Ci/MWt) [DIN # 10]	Gap Fraction [Table 3]	Power* (Fraction of Rods Failed)* (Peaking Factor) (MWt)	Pool DF[a] [§3.17.2, §3.17.3]	FHA Release Activity (Ci)
1	BR 82	1.2390E+02	5%	18.029	200	5.5845E-01
2	BR 83	3.2960E+00	5%	18.029	200	1.4856E-02
3	KR 83M	1.2750E+01	5%	18.029	1	1.1494E+01
4	KR 85	4.1550E+02	10%	18.029	1	7.4911E+02
5	KR 85M	1.6560E+02	5%	18.029	1	1.4928E+02
6	KR 87	2.6830E-02	5%	18.029	1	2.4186E-02
7	KR 88	5.1150E+01	5%	18.029	1	4.6109E+01
8	RB 87	N/A[b]	~~5%~~ 12%	18.029	∞	0
9	RB 88	5.7120E+01[c]	~~5%~~ 12%	18.029	∞	0
10	I129	1.3910E-03	5%	18.029	200	6.2696E-06
11	I130	3.0390E+02	5%	18.029	200	1.3698E+00
12	I131	2.5290E+04	8%	18.029	200	1.8238E+02
13	I132	3.2140E+04	5%	18.029	200	1.4486E+02
14	I133	2.5280E+04	5%	18.029	200	1.1394E+02
15	I134	1.3320E-03	5%	18.029	200	6.0037E-06
16	I135	4.1600E+03	5%	18.029	200	1.8750E+01
17	XE129M	2.3050E-01	5%	18.029	1	2.0778E-01
18	XE131M	3.0440E+02	5%	18.029	1	2.7440E+02
19	XE133	5.1070E+04	5%	18.029	1	4.6037E+04
20	XE133M	1.5560E+03	5%	18.029	1	1.4027E+03
21	XE135	1.4060E+04	5%	18.029	1	1.2674E+04
22	XE135M	6.6640E+02	5%	18.029	1	6.0073E+02
23	CS135	2.7050E-02[c]	~~5%~~ 12%	18.029	∞	0

AW 1-7-02

AW 1-7-02

Notes:

a. Note that the infinite value for pool DF of RB 87, RB 88, and CS135 indicates that cesium and rubidium were assumed to retain completely in the pool (§3.17.3).

b. DIN # 10 does not list the core inventory for RB 87.

c. The core inventory of RB 88 and CS135 are *not* used for the calculation because of infinite pool DF and are listed only for information. [DIN # 10]

Table 6. Radionuclides Decay Data Used for the Analysis [DIN # 15]

No.	Isotope	Daughter 1		Daughter 2		Radioactive Half-life, $t_{1/2}$	
		Nuclide	Fraction	Nuclide	Fraction	-	(s)
1	BR 82					35.3 h	1.2708000E+05
2	BR 83	KR 83M	1			2.39 h	8.6040000E+03
3	KR 83M					1.83 h	6.5880000E+03
4	KR 85					10.72 y	3.3806592E+08
5	KR 85M	KR 85	0.211			4.48 h	1.6128000E+04
6	KR 87	RB 87	1			76.3 m	4.5780000E+03
7	KR 88	RB 88	1			2.84 h	1.0224000E+04
8	RB 87					4.70E+10 y	1.4821920E+18
9	RB 88					17.8 m	1.0680000E+03
10	I129					1.57E+07 y	4.9511520E+14
11	I130					12.36 h	4.4496000E+04
12	I131	XE131M	0.0111			8.04 d	6.9465600E+05
13	I132					2.3 h	8.2800000E+03
14	I133	XE133M	0.029	XE133	0.971	20.8 h	7.4880000E+04
15	I134					52.6 m	3.1560000E+03
16	I135	XE135M	0.154	XE135	0.846	6.61 h	2.3796000E+04
17	XE129M					8 d	6.9120000E+05
18	XE131M					11.9 d	1.0281600E+06
19	XE133					5.245 d	4.5316800E+05
20	XE133M	XE133	1			2.188 d	1.8904320E+05
21	XE135	Cs-135	1			9.09 h	3.2724000E+04
22	XE135M	XE135	0.9999	CS135	4.50E-05	15.29 m	9.1740000E+02
23	CS135					2.30E+06 y	7.2532800E+13

Table 7. Comparison of TEDE Calculated for Control Room, EAB, and LPZ with Regulatory Limits

Location	Control Room	EAB	LPZ
RADTRAD results (rem) [Attachment 5]	1.03	1.44	0.161
Regulatory limit (rem) [§2]	5	6.3	6.3
$\frac{\text{RADTRAD Value}}{\text{Regulatory Limit}}$	20.5%	22.9%	2.6%

4.4 Sensitivity Analysis

4.4.1 Sensitivity Case 1: Effect of CR Isolation and Fresh Air Intake

Sensitivity Case 1 was run to study the effect of control room isolation and fresh air intake. The Rad Monitor was assumed to isolate the air intake at the worst possible time once the available activity is introduced into the control room. This is considered conservative as it maximizes the ~~does~~ DOSE to the control room operators. No inleakage of air into the control room was assumed. This is conservative, as additional inleakage of fresh air would tend to dilute the radioactivity in the control room. At 2 hours, outside air purge at a rate of 5400cfm (6000cfm -10%) assumed to initiate and continue until the end of dose analysis (30 days). [DIN # 26, 27] The 5400 cfm was considered to be conservative as it kept the activity in the control room longer. AW 1-7-02

The RADTRAD computer code was run with the main input file, which was changed to reflect the above flow rate data, and the same auxiliary input files that were used for the base case. The computer output is listed in Attachment 6. Table 8 compares the control room TEDE for this case with the base case and sensitivity case 2 results. The control room TEDE calculated for starting fresh air intake after 2 hours following control room isolation is 2.81 rem, which is below and 56% of the regulatory limit.

4.4.2 Sensitivity Case 2: Effect of CR Isolation and Recirculation Filtering

Sensitivity Case 2 was run to study the effect of control room isolation and Recirculation Filtering. The Rad Monitor was assumed to isolate the air intake at the worst possible time once the available activity is introduced into the control room. No inleakage of air into the control room was assumed. This is conservative, as additional inleakage of fresh air would tend to dilute the radioactivity in the control room. At 2 hours, the control room emergency recirculation was assumed to initiate. A recirculation flow rate of 27,000 cfm (30,000 -10%) and a charcoal efficiency of 50% were chosen to be consistent with assumptions in the LOCA analysis. [DIN # 18]

The RADTRAD computer code was run with the main input file, which was changed to reflect the above flow rate data, and the same auxiliary input files that were used for the base case. The computer output is listed in Attachment 7. Table 8 compares the control room TEDE for this case with the base case~~s~~ and sensitivity case 1 results. The control room TEDE calculated for starting recirculation filtering after 2 hours following control room isolation is 2.97 rem, which is below and 59% of the regulatory limit. AW 1-7-02

Table 8. Comparison of Sensitivity Analysis Results with the Base Case for TEDE Calculated for Control Room

Case	Control Room TEDE (rem)
Base Case [Table 7]	1.03
Sensitivity Case 1: Effect of CR Isolation and Fresh Air Intake [Attachment 6]	2.81
Sensitivity Case 2: Effect of CR Isolation and Recirculation Filtering [Attachment 7]	2.97

5 COMPUTER INPUT AND OUTPUT

The main input file for the base case, pnpp_fha.psf, and the three auxiliary input files, pnpp_fha.rft, pnpp_fha.nif, and pnpp_fha.dcf were used as input to the code. These files are given in Attachment 1 through Attachment 4. The output file, pnpp_fha.out, is given in Attachment 5. Computer output for sensitivity cases 1 and 2 are given in Attachment 6 and Attachment 7.

Appendix A. Fuel Handling Accident while Transiting over the Refueling Shield

Once a fuel bundle is removed from the core, it is typically taken to either the Inclined Fuel Transfer Tube or the Upper Containment Fuel Pool racks located in an adjacent pool. In order to accomplish this the fuel must be moved from the Reactor Cavity over the Refueling Shield into the adjacent pool. The refueling shield is a device that is put in physically placed during a refueling outage in order to provide radiological shielding to the drywell area below the reactor cavity. The bottom of the refueling shield contains 8"of lead sandwiched between stainless steel plates. The shield is set onto locating pins on the Reactor Cavity/Steam Dryer gate opening and sits on the reactor flange.

In performing reviews of the License Amendment Request for Fuel Handling Accident re-analysis, it was determined that the current Technical Specification Bases for the amount of water above the reactor flange during movement of irradiated fuel within the RPV may not have considered all potential fuel handling accident scenarios.

Currently, Technical Specification 3.9.6 require a minimum water level of 22'-9" above the reactor flange during movement of irradiated fuel within the RPV. [DIN # 23] Regulatory Guidance 1.183 specifies a requirement for 23' providing the basis for the iodine decontamination factor used in the analysis. [DIN # 2] The Tech Spec Bases provides an assessment that while the worst case assumption include the dropping of the irradiated fuel assembly onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. The Bases goes on to say that dropping an assembly on the RPV flange will result in reduced releases of fission gases. The Bases conclude that the operation with slightly less than 23' is acceptable in the event of a fuel drop on the reactor flange. The NRC in their response to Perry letter PY-CEI/NRR-1510L had accepted this. [DIN # 29] Perry's letter further noted shielding >23' over the Upper Containment Pool fuel racks and Inclined Fuel Transfer System Upender.

The amount of shielding assumed in the current Technical Specification Section 3.9.6 is 22'-9".
The Refueling Shield bottom height is 9.25" (0.75" thick bottom plate, 8.0" thickness of lead, 0.5" thick upper plate). (Perry Drawing 4549-0194-1) [DIN # 17] Fuel bundle channel square dimension is 5.72" (Perry Drawing 4549-18-1) [DIN # 17]

Therefore, the least amount of shielding for a dropped bundle on the Refueling Shield is ~ 21'6" (21.5') (=22.75' - ((9.25" + 5.72")/12)). This is less than the 22'-9" assumed in the Technical Specifications.

The purpose of this appendix is to examine the effect of this reduced water level on radiological doses at control room, EAB, and LPZ.

Overall decontamination factor for halogen species, DF, can be calculated using individual decontamination factors for inorganic and organic fractions of the species as given in Equation 10.

$$DF = \frac{1}{\frac{f_{inorg}}{DF_{inorg}} + \frac{f_{org}}{DF_{org}}} \qquad \text{Equation 10}$$

Equation 10 can be rewritten to obtain the decontamination factor for inorganic halogen species for known overall and organic decontamination factors as given in Equation 11.

$$DF_{inorg} = \frac{f_{inorg}}{\frac{1}{DF} - \frac{f_{org}}{DF_{org}}} \qquad \text{Equation 11}$$

Burley calculated the decontamination factor for inorganic (elemental) iodine as given by Equation 12. [DIN # 30]

$$DF_{inorg} = \exp\left(\frac{6}{d_b} k_{eff} \frac{H}{v_b}\right) \qquad \text{Equation 12}$$

Where

d_b - Bubble diameter

H - Bubble rise height, i.e., the effective depth of water, defined as the water depth between the top of the damaged fuel rods and the fuel pool surface

k_{eff} - Mass transfer coefficient

v_b - Bubble velocity

Equation 12 can be rewritten as Equation 13 using a constant, C, as defined in Equation 14.

$$DF_{inorg} = \exp(CH) \qquad \text{Equation 13}$$

$$C = \frac{6}{d_b} k_{eff} \frac{1}{v_b} \qquad \text{Equation 14}$$

Using Equation 13, the decontamination factor for inorganic halogen species for a given pool depth can be expressed as given in Equation 15.

$$DF_{inorg,0} = \exp(CH_0) \qquad \text{Equation 15}$$

Equation 16 is obtained by substituting for C from Equation 15 in Equation 13.

$$DF_{inorg} = \left(DF_{inorg,0}\right)^{\frac{H}{H_0}} \qquad \text{Equation 16}$$

Calculations for the design basis FHA, described in the main report, assumed an overall decontamination factor of 200 for a pool depth of 23', and inorganic and organic fractions of halogen species of 99.85% and 0.15% (§3.17.1.3 and 3.17.2). Substituting these values in Equation 11, the corresponding decontamination factor for inorganic halogen species was calculated as 285.3. Using this value for a pool depth of 23', the decontamination factor for inorganic halogen species for a reduced pool depth of 21.5' was calculated as 197.3, using Equation 16. The corresponding value of overall decontamination factor for halogen species was calculated as 152.4, using Equation 10, which is 76.2% of the overall decontamination factor of 200 used in the main report.

Calculations for the design basis FHA, described in the main report, assumed a total number of fuel rods damaged during an FHA as 151 (§3.17.1.1). However, the number of fuel rods damaged in for an FHA occurring while transiting over the Refueling Shield would be limited to 85.84, which is the total number of equivalent full-length fuel rods in a fuel assembly. [DIN # 12] Therefore, the number of fuel rods damaged FHA while transiting over the Refueling Shield is 56.8% (=85.84/151) of that assumed for the design basis FHA.

The doses were calculated to be 74.5% (= (56.8%)/(76.2%)) of those calculated for the design basis FHA. Therefore, the dose consequences for an FHA occurring while transiting over the Refueling Shield will be bounded by the dose consequences for the design basis FHA.

```
Radtrad 3.02 1/5/2000
perry fha
Nuclide Inventory File:
d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.nif
Plant Power Level:
 3.8332E+03
Compartments:
  3
Compartment 1:
Containment
  3
 1.0000E+00
  0
  0
  0
  0
  0
Compartment 2:
Environment
  2
 0.0000E+00
  0
  0
  0
  0
  0
Compartment 3:
Control Room
  1
 3.6707E+05
  0
  0
  0
  0
  0
Pathways:
  3
Pathway 1:
Unfiltered Release to Environment
  1
  2
  4
Pathway 2:
Unfiltered Environment to CR
  2
  3
  2
Pathway 3:
Control Room Exhaust
  3
  2
  2
End of Plant Model File
Scenario Description Name:

Plant Model Filename:
```

```
Source Term:
  1
  1   1.0000E+00
d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.dcf
d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.rft
  0.0000E+00
  1
  0.0000E+00   0.9985E+00   0.0015E+00   1.0000E+00
Overlying Pool:
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
  3
Compartment 1:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 2:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 3:
  1
  1
  0
  0
  0
  0
  0
  0
  0
Pathways:
  3
Pathway 1:
  0
  0
  0
  0
  0
  0
  0
```

```
  0
  0
  0
  1
  2
  0.0000E+00   1.0000E+10
  7.2000E+02   0.0000E+00
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0.0001E+00   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 3:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0.0001E+00   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
Exclusion Area Boundary
  2
  1
  2
  0.0000E+00   4.3000E-04
  2.0000E+00   0.0000E+00
  1
  3
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  0
```

```
Location 2:
Outer Boundary of the LPZ
  2
  1
  2
  0.0000E+00   4.8000E-05
  2.0000E+00   0.0000E+00
  1
  3
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  0
Location 3:
Control Room
  3
  0
  1
  2
  0.0000E+00   3.5000E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Effective Volume Location:
  1
  2
  0.0000E+00   3.5000E-04
  2.0000E+00   0.0000E+00
Simulation Parameters:
  4
  0.0000E+00   2.5000E-02
  8.0000E+00   1.0000E-01
  2.4000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Output Filename:

  1
  1
  1
  0
  0
End of Scenario File
```

```
Release Fraction and Timing Name:
Perry FHA
Duration (h):
 0.0001E+00  0.0000E+00  0.0000E+00  0.0000E+00
Noble Gases:
 0.1000E+01  0.0000E+00  0.0000E+00  0.0000E+00
Iodine:
 0.1000E+01  0.0000E+00  0.0000E+00  0.0000E+00
Cesium:
 0.1000E+01  0.0000E+00  0.0000E+00  0.0000E+00
Tellurium:
 0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
Strontium:
 0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
Barium:
 0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
Ruthenium:
 0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
Cerium:
 0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
Lanthanum:
 0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
Non-Radioactive Aerosols (kg):
 0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
End of Release File
```

```
Nuclide Inventory Name:
 Perry FHA
 Power Level:
  3.8332E+03
 Nuclides:
  23
 Nuclide 001:
 Br-82
   2
  1.2708000000E+05
  0.8200E+02
  5.5845E-01
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 002:
 Br-83
   2
  8.6040000000E+03
  0.8300E+02
  1.4856E-02
 Kr-83m    0.1000E+01
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 003:
 Kr-83m
   1
  6.5880000000E+03
  0.8300E+02
  1.1494E+01
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 004:
 Kr-85
   1
  3.3806592000E+08
  0.8500E+02
  7.4911E+02
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 005:
 Kr-85m
   1
  1.6128000000E+04
  0.8500E+02
  1.4928E+02
 Kr-85     0.2110E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 006:
 Kr-87
   1
  0.4578000000E+04
  0.8700E+02
  2.4186E-02
 Rb-87     0.1000E+01
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 007:
 Kr-88
   1
  1.0224000000E+04
  0.8800E+02
  4.6109E+01
 Rb-88     0.1000E+01
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 008:
 Rb-87
   3
  1.4821920000E+18
  0.8700E+02
  0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 009:
 Rb-88
   3
  1.0680000000E+03
  0.8800E+02
  0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 010:
 I-129
   2
  4.9511520000E+14
  0.1290E+03
  6.2696E-06
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 011:
 I-130
   2
  4.4496000000E+04
  0.1300E+03
  1.3698E+00
 none      0.0000E+00
 none      0.0000E+00
 none      0.0000E+00
 Nuclide 012:
 I-131
   2
  6.9465600000E+05
  0.1310E+03
  1.8238E+02
 Xe-131m   0.1110E-01
 none      0.0000E+00
 none      0.0000E+00
```

```
Nuclide 013:
I-132
  2
 8.2800000000E+03
 0.1320E+03
 1.4486E+02
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 014:
I-133
  2
 7.4880000000E+04
 0.1330E+03
 1.1394E+02
Xe-133m   0.2900E-01
Xe-133    0.9710E+00
none      0.0000E+00
Nuclide 015:
I-134
  2
 0.3156000000E+04
 0.1340E+03
 6.0037E-06
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 016:
I-135
  2
 2.3796000000E+04
 0.1350E+03
 1.8750E+01
Xe-135m   0.1540E+00
Xe-135    0.8460E+00
none      0.0000E+00
Nuclide 017:
Xe-129m
  1
 6.9120000000E+05
 0.1290E+03
 2.0778E-01
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 018:
Xe-131m
  1
 1.0281600000E+06
 0.1310E+03
 2.7440E+02
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 019:
Xe-133
  1
 4.5316800000E+05
 0.1330E+03
 4.6037E+04
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
Nuclide 020:
Xe-133m
  1
 1.8904320000E+05
 0.1330E+03
 1.4027E+03
Xe-133    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 021:
Xe-135
  1
 3.2724000000E+04
 0.1350E+03
 1.2674E+04
Cs-135    0.1000E+01
none      0.0000E+00
none      0.0000E+00
Nuclide 022:
Xe-135m
  1
 9.1740000000E+02
 0.1350E+03
 6.0073E+02
Xe-135    0.9999E+00
Cs-135    4.5000E-05
none      0.0000E+00
Nuclide 023:
Cs-135
  3
 7.2532800000E+13
 0.1350E+03
 0.0000E+00
none      0.0000E+00
none      0.0000E+00
none      0.0000E+00
End of Nuclear Inventory File
```

fgr 11 and 12 dcfs

9 ORGANS DEFINED IN THIS FILE:
GONADS
BREAST
LUNGS
RED MARR
BONE SUR
THYROID
REMAINDER
EFFECTIVE
SKIN(FGR)

23 NUCLIDES DEFINED IN THIS FILE:

Br-82	D
Br-83	H
Kr-83m	
Kr-85	
Kr-85m	
Kr-87	
Kr-88	
Rb-87	Y
Rb-88	
I-129	Y
I-130	H
I-131	D
I-132	D
I-133	D
I-134	M
I-135	D
Xe-129m	
Xe-131m	
Xe-133	
Xe-133m	
Xe-135	
Xe-135m	
Cs-135	Y

	CLOUDSHINE	GROUND SHINE 8HR	GROUND SHINE 7DAY	GROUND SHINE RATE	INHALED ACUTE	INHALED CHRONIC	INGESTION
Br-82							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.300E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.130E-10	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Br-83							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	3.820E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.410E-11	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Kr-83m							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.500E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Kr-85							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.190E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Kr-85m							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	7.480E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Kr-87							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	4.120E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Kr-88							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.020E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Rb-87							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.820E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00	8.740E-10	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Rb-88							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

EFFECTIVE	3.360E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.260E-11	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-129							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	3.800E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.690E-08	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-130							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.040E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.140E-10	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-131							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.820E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	8.890E-09	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-132							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.120E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.030E-10	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-133							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	2.940E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.580E-09	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-134							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.300E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.550E-11	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-135							

GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	8.294E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.320E-10	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Xe-129m							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.060E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Xe-131m							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	3.890E-16	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Xe-133							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.560E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Xe-133m							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.370E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Xe-135							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	1.190E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Xe-135m							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	2.040E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cs-135							
GONADS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BREAST	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LUNGS	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RED MARR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BONE SUR	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
THYROID	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
REMAINDER	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EFFECTIVE	5.650E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.230E-09	0.000E+00
SKIN(FGR)	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

```
#######################################################################
          RADTRAD Version 3.02 run on 11/15/2001  at 16:53:48
#######################################################################

#######################################################################
                    File information
#######################################################################

 Plant file name            = pnpp_fha.psf
 Inventory file name        = d:\hwagage\computer codes\radtrad\run
batch\perry\pnpp_fha.nif
 Scenario file name         = NEW_SDF.SDF
 Release file name          = d:\hwagage\computer codes\radtrad\run
batch\perry\pnpp_fha.rft
 Dose conversion file name = d:\hwagage\computer codes\radtrad\run
batch\perry\pnpp_fha.dcf


     #####    ####   #####    #  #     #  #####   #    #  #####
     #    #  #       #        #  ##    #  #    #  #    #    #
     #    #  #       #        #  # #   #  #    #  #    #    #
     #####    ####   ####     #  #  #  #  #####   #    #    #
     #            #  #        #  #   # #  #       #    #    #
     #            #  #        #  #    ##  #       #    #    #
     #        ####   #        #  #     #  #        ####     #


 Radtrad 3.02 1/5/2000
 perry fha
 Nuclide Inventory File:
 d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.nif
 Plant Power Level:
  3.8332E+03
 Compartments:
   3
 Compartment 1:
 Containment
   3
  1.0000E+00
   0
   0
   0
   0
   0
 Compartment 2:
 Environment
   2
  0.0000E+00
   0
   0
   0
   0
   0
 Compartment 3:
 Control Room
   1
  3.6707E+05
   0
   0
   0
   0
   0
```

```
Pathways:
  3
Pathway 1:
Unfiltered Release to Environment
  1
  2
  4
Pathway 2:
Unfiltered Environment to CR
  2
  3
  2
Pathway 3:
Control Room Exhaust
  3
  2
  2
End of Plant Model File
Scenario Description Name:

Plant Model Filename:

Source Term:
  1
  1   1.0000E+00
d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.dcf
d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.rft
  0.0000E+00
  1
  0.0000E+00   0.9985E+00   0.0015E+00   1.0000E+00
Overlying Pool:
  0
  0.0000E+00
  0
  0
  0
  0
Compartments:
  3
Compartment 1:
  0
  1
  0
  0
  0
  0
  0
  0
  0
Compartment 2:
  0
  1
  0
  0
  0
  0
  0
  0
  0
```

```
Compartment 3:
  1
  1
  0
  0
  0
  0
  0
  0
  0
Pathways:
  3
Pathway 1:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  2
  0.0000E+00   1.0000E+10
  7.2000E+02   0.0000E+00
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0.0001E+00   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 3:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0.0001E+00   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
```

```
Dose Locations:
  3
Location 1:
Exclusion Area Boundary
  2
  1
  2
  0.0000E+00   4.3000E-04
  2.0000E+00   0.0000E+00
  1
  3
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  0
Location 2:
Outer Boundary of the LPZ
  2
  1
  2
  0.0000E+00   4.8000E-05
  2.0000E+00   0.0000E+00
  1
  3
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  0
Location 3:
Control Room
  3
  0
  1
  2
  0.0000E+00   3.5000E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Effective Volume Location:
  1
  2
  0.0000E+00   3.5000E-04
  2.0000E+00   0.0000E+00
Simulation Parameters:
  4
  0.0000E+00   2.5000E-02
  8.0000E+00   1.0000E-01
  2.4000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Output Filename:

  1
  1
  1
  0
  0
End of Scenario File
```

```
#######################################################################
          RADTRAD Version 3.02 run on 11/15/2001  at 16:53:48
#######################################################################

#######################################################################
                    Plant Description
#######################################################################

Number of Nuclides =  23

Inventory Power =   3.8332E+03 MWth
Plant Power Level =   3.8332E+03 MWth

Number of compartments    =   3

Compartment information

Compartment number  1  (Source term fraction =   1.0000E+00
)
Name: Containment
Compartment volume =   1.0000E+00 (Cubic feet)
Pathways into and out of compartment  1
    Pathway to compartment number   2: Unfiltered Release to Environment

Compartment number  2
Name: Environment
Pathways into and out of compartment  2
    Pathway to compartment number   3: Unfiltered Environment to CR
    Pathway from compartment number 1: Unfiltered Release to Environment
    Pathway from compartment number 3: Control Room Exhaust

Compartment number  3
Name: Control Room
Compartment volume =   3.6707E+05 (Cubic feet)
Pathways into and out of compartment  3
    Pathway to compartment number   2: Control Room Exhaust
    Pathway from compartment number 2: Unfiltered Environment to CR

Total number of pathways =   3
```

```
#######################################################################
          RADTRAD Version 3.02 run on 11/15/2001  at 16:53:48
#######################################################################

#######################################################################
                     Scenario Description
#######################################################################

Radioactive Decay is enabled
Calculation of Daughters is enabled
RELEASE_NAME = Perry FHA
Release Fractions and Timings
                GAP        EARLY IN-VESSEL
             0.0001 hrs     0.0000 hrs
NOBLES       1.0000E+00     0.0000E+00
IODINE       1.0000E+00     0.0000E+00
CESIUM       1.0000E+00     0.0000E+00
TELLURIUM    0.0000E+00     0.0000E+00
STRONTIUM    0.0000E+00     0.0000E+00
BARIUM       0.0000E+00     0.0000E+00
RUTHENIUM    0.0000E+00     0.0000E+00
CERIUM       0.0000E+00     0.0000E+00
LANTHANUM    0.0000E+00     0.0000E+00

Iodine fractions
     Aerosol      =   0.0000E+00
     Elemental    =   9.9850E-01
     Organic      =   1.5000E-03

COMPARTMENT DATA

Compartment number   1: Containment

Compartment number   2: Environment

Compartment number   3: Control Room

PATHWAY DATA

Pathway number  1: Unfiltered Release to Environment

     Convection Data
      Time (hr)          Flow Rate (% / day)
       0.0000E+00              1.0000E+10
       7.2000E+02              0.0000E+00

Pathway number  2: Unfiltered Environment to CR

     Pathway Filter: Removal Data

       Time (hr)       Flow Rate          Filter Efficiencies (%)
                        (cfm)        Aerosol     Elemental     Organic
       0.0000E+00     6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
       1.0000E-04     5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
       7.2000E+02     5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00

Pathway number  3: Control Room Exhaust

     Pathway Filter: Removal Data

       Time (hr)       Flow Rate          Filter Efficiencies (%)
                        (cfm)        Aerosol     Elemental     Organic
```

0.0000E+00	6.6000E+03	0.0000E+00	0.0000E+00	0.0000E+00
1.0000E-04	5.4000E+03	0.0000E+00	0.0000E+00	0.0000E+00
7.2000E+02	5.4000E+03	0.0000E+00	0.0000E+00	0.0000E+00

LOCATION DATA

Location Exclusion Area Boundary is in compartment 2

Location X/Q Data

Time (hr)	X/Q (s * m^-3)
0.0000E+00	4.3000E-04
2.0000E+00	0.0000E+00

Location Breathing Rate Data

Time (hr)	Breathing Rate (m^3 * sec^-1)
0.0000E+00	3.5000E-04
8.0000E+00	1.8000E-04
2.4000E+01	2.3000E-04

Location Outer Boundary of the LPZ is in compartment 2

Location X/Q Data

Time (hr)	X/Q (s * m^-3)
0.0000E+00	4.8000E-05
2.0000E+00	0.0000E+00

Location Breathing Rate Data

Time (hr)	Breathing Rate (m^3 * sec^-1)
0.0000E+00	3.5000E-04
8.0000E+00	1.8000E-04
2.4000E+01	2.3000E-04

Location Control Room is in compartment 3

Location X/Q Data

Time (hr)	X/Q (s * m^-3)
0.0000E+00	3.5000E-04
2.0000E+00	0.0000E+00

Location Breathing Rate Data

Time (hr)	Breathing Rate (m^3 * sec^-1)
0.0000E+00	3.5000E-04
7.2000E+02	0.0000E+00

Location Occupancy Factor Data

Time (hr)	Occupancy Factor
0.0000E+00	1.0000E+00
2.4000E+01	6.0000E-01
9.6000E+01	4.0000E-01
7.2000E+02	0.0000E+00

USER SPECIFIED TIME STEP DATA - SUPPLEMENTAL TIME STEPS

Time	Time step
0.0000E+00	2.5000E-02
8.0000E+00	1.0000E-01
2.4000E+01	4.0000E-01
7.2000E+02	0.0000E+00

```
#######################################################################
          RADTRAD Version 3.02 run on 11/15/2001  at 16:53:48
#######################################################################

 ####   #    #  #####  #####   #    #  #####
#    # #    #    #    #    #  #    #    #
#    # #    #    #    #    #  #    #    #
#    # #    #    #    #####   #    #    #
#    # #    #    #    #       #    #    #
#    # #    #    #    #       #    #    #
 ####    ####     #    #        ####     #

#######################################################################
           Dose, Detailed model and Detailed Inventory Output
#######################################################################
```

Exclusion Area Boundary Doses:

Time (h) = 0.0001	Whole Body	Thyroid	TEDE
Delta dose (rem)	4.2456E-01	0.0000E+00	1.4377E+00
Accumulated dose (rem)	4.2456E-01	0.0000E+00	1.4377E+00

Outer Boundary of the LPZ Doses:

Time (h) = 0.0001	Whole Body	Thyroid	TEDE
Delta dose (rem)	4.7393E-02	0.0000E+00	1.6048E-01
Accumulated dose (rem)	4.7393E-02	0.0000E+00	1.6048E-01

Control Room Doses:

Time (h) = 0.0001	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.2078E-06	0.0000E+00	4.5686E-05
Accumulated dose (rem)	1.2078E-06	0.0000E+00	4.5686E-05

Control Room Compartment Nuclide Inventory:

Time (h) = 0.0001	Ci	kg	Atoms	Bq
Br-82	6.0733E-04	5.6097E-13	4.1198E+12	2.2471E+07
Br-83	1.6156E-05	1.0227E-15	7.4200E+09	5.9776E+05
Kr-83m	1.2500E-02	6.0583E-13	4.3957E+12	4.6248E+08
Kr-85	8.1467E-01	2.0751E-06	1.4702E+19	3.0143E+10
Kr-85m	1.6234E-01	1.9727E-11	1.3976E+14	6.0067E+09
Kr-87	2.6301E-05	9.2854E-16	6.4273E+09	9.7315E+05
Kr-88	5.0143E-02	3.9989E-12	2.7366E+13	1.8553E+09
Rb-87	4.4281E-24	5.0613E-20	3.5034E+05	1.6384E-13
Rb-88	1.1716E-05	9.7601E-17	6.6792E+08	4.3349E+05
I-129	6.8183E-09	3.8601E-08	1.8020E+17	2.5228E+02
I-130	1.4897E-03	7.6381E-13	3.5383E+12	5.5118E+07
I-131	1.9834E-01	1.5999E-09	7.3546E+15	7.3387E+09
I-132	1.5753E-01	1.5262E-11	6.9627E+13	5.8288E+09
I-133	1.2391E-01	1.0938E-10	4.9529E+14	4.5847E+09
I-134	6.5286E-09	2.4473E-19	1.0999E+06	2.4156E+02
I-135	2.0391E-02	5.8063E-12	2.5901E+13	7.5446E+08
Xe-129m	2.2597E-04	1.7859E-12	8.3372E+12	8.3607E+06
Xe-131m	2.9842E-01	3.5627E-09	1.6378E+16	1.1041E+10
Xe-133	5.0066E+01	2.6747E-07	1.2111E+18	1.8525E+12
Xe-133m	1.5255E+00	3.3997E-09	1.5394E+16	5.6442E+10

Xe-135	1.3783E+01	5.3973E-09	2.4076E+16	5.0998E+11
Xe-135m	6.5313E-01	7.1700E-12	3.1984E+13	2.4166E+10
Cs-135	4.7418E-14	4.1157E-14	1.8359E+11	1.7545E-03

Control Room Transport Group Inventory:

Time (h) = 0.0001	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5969E+19	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.8787E+17	0.0000E+00	0.0000E+00
Organic I (atoms)	2.8223E+14	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 0.0001	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 0.0001	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 0.0001	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 2.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.3779E-03	0.0000E+00	4.7275E-03
Accumulated dose (rem)	4.2594E-01	0.0000E+00	1.4424E+00

Outer Boundary of the LPZ Doses:

Time (h) = 2.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.5382E-04	0.0000E+00	5.2772E-04
Accumulated dose (rem)	4.7547E-02	0.0000E+00	1.6101E-01

Control Room Doses:

Time (h) = 2.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	2.1084E-02	0.0000E+00	8.5359E-01
Accumulated dose (rem)	2.1086E-02	0.0000E+00	8.5363E-01

Control Room Compartment Nuclide Inventory:

Time (h) = 2.0000	Ci	kg	Atoms	Bq
Br-82	1.0029E-04	9.2638E-14	6.8034E+11	3.7108E+06
Br-83	1.5536E-06	9.8342E-17	7.1353E+08	5.7483E+04
Kr-83m	1.0076E-03	4.8836E-14	3.5434E+11	3.7281E+07
Kr-85	1.3992E-01	3.5639E-07	2.5250E+18	5.1771E+09

Kr-85m	2.0462E-02	2.4865E-12	1.7616E+13	7.5711E+08
Kr-87	1.5187E-06	5.3615E-17	3.7112E+08	5.6191E+04
Kr-88	5.2861E-03	4.2156E-13	2.8849E+12	1.9559E+08
Rb-87	9.2626E-21	1.0587E-16	7.3285E+08	3.4272E-10
Rb-88	5.9844E-03	4.9854E-14	3.4117E+11	2.2142E+08
I-129	1.1711E-09	6.6298E-09	3.0950E+16	4.3329E+01
I-130	2.2871E-04	1.1727E-13	5.4323E+11	8.4623E+06
I-131	3.3822E-02	2.7281E-10	1.2541E+15	1.2514E+09
I-132	1.4809E-02	1.4347E-12	6.5453E+12	5.4793E+08
I-133	1.9910E-02	1.7576E-11	7.9582E+13	7.3667E+08
I-135	2.8396E-03	8.0858E-13	3.6069E+12	1.0507E+08
Xe-129m	3.8531E-05	3.0453E-13	1.4216E+12	1.4256E+06
Xe-131m	5.1007E-02	6.0896E-10	2.7994E+15	1.8873E+09
Xe-133	8.5079E+00	4.5453E-08	2.0581E+17	3.1479E+11
Xe-133m	2.5519E-01	5.6873E-10	2.5752E+15	9.4421E+09
Xe-135	2.0357E+00	7.9713E-10	3.5559E+15	7.5319E+10
Xe-135m	9.5488E-04	1.0483E-14	4.6761E+10	3.5331E+07
Cs-135	1.5125E-10	1.3128E-10	5.8561E+14	5.5963E+00

Control Room Transport Group Inventory:

Time (h) = 2.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	2.7427E+18	0.0000E+00	0.0000E+00
Elemental I (atoms)	3.2268E+16	0.0000E+00	0.0000E+00
Organic I (atoms)	4.8474E+13	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 2.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 2.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 2.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 8.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2594E-01	0.0000E+00	1.4424E+00

Outer Boundary of the LPZ Doses:

Time (h) = 8.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00

Accumulated dose (rem)	4.7547E-02	0.0000E+00	1.6101E-01

Control Room Doses:

Time (h) = 8.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	3.6824E-03	0.0000E+00	1.7162E-01
Accumulated dose (rem)	2.4768E-02	0.0000E+00	1.0253E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 8.0000	Ci	kg	Atoms	Bq
Br-82	4.4677E-07	4.1267E-16	3.0307E+09	1.6530E+04
Br-83	1.3664E-09	8.6496E-20	6.2758E+05	5.0558E+01
Kr-83m	5.2275E-07	2.5337E-17	1.8383E+08	1.9342E+04
Kr-85	7.0120E-04	1.7860E-09	1.2654E+16	2.5944E+07
Kr-85m	4.0529E-05	4.9249E-15	3.4892E+10	1.4996E+06
Kr-88	6.1254E-06	4.8850E-16	3.3430E+09	2.2664E+05
Rb-87	6.9036E-23	7.8908E-19	5.4620E+06	2.5543E-12
Rb-88	7.0418E-06	5.8663E-17	4.0145E+08	2.6055E+05
I-129	5.8690E-12	3.3226E-11	1.5511E+14	2.1715E-01
I-130	8.1872E-07	4.1978E-16	1.9446E+09	3.0293E+04
I-131	1.6589E-04	1.3381E-12	6.1513E+12	6.1379E+06
I-132	1.2168E-05	1.1788E-15	5.3779E+09	4.5021E+05
I-133	8.1699E-05	7.2121E-14	3.2656E+11	3.0229E+06
I-135	7.5856E-06	2.1600E-15	9.6354E+09	2.8067E+05
Xe-129m	1.8897E-07	1.4935E-15	6.9721E+09	6.9917E+03
Xe-131m	2.5196E-04	3.0081E-12	1.3828E+13	9.3226E+06
Xe-133	4.1296E-02	2.2062E-10	9.9894E+14	1.5279E+09
Xe-133m	1.1817E-03	2.6337E-12	1.1925E+13	4.3725E+07
Xe-135	6.4602E-03	2.5297E-12	1.1285E+13	2.3903E+08
Xe-135m	1.2568E-06	1.3797E-17	6.1547E+07	4.6502E+04
Cs-135	2.4484E-12	2.1251E-12	9.4797E+12	9.0591E-02

Control Room Transport Group Inventory:

Time (h) = 8.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.3745E+16	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.6171E+14	0.0000E+00	0.0000E+00
Organic I (atoms)	2.4294E+11	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 8.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 8.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 8.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00

Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 24.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2594E-01	0.0000E+00	1.4424E+00

Outer Boundary of the LPZ Doses:

Time (h) = 24.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7547E-02	0.0000E+00	1.6101E-01

Control Room Doses:

Time (h) = 24.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.3641E-05	0.0000E+00	8.2602E-04
Accumulated dose (rem)	2.4782E-02	0.0000E+00	1.0261E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 24.0000	Ci	kg	Atoms	Bq
Kr-85	5.1571E-10	1.3136E-15	9.3064E+09	1.9081E+01
I-129	4.3169E-18	2.4440E-17	1.1409E+08	1.5973E-07
I-131	1.1520E-10	9.2926E-19	4.2718E+06	4.2626E+00
I-133	3.5259E-11	3.1125E-20	1.4093E+05	1.3046E+00
Xe-131m	1.7832E-10	2.1289E-18	9.7868E+06	6.5979E+00
Xe-133	2.7883E-08	1.4896E-16	6.7449E+08	1.0317E+03
Xe-133m	7.0400E-10	1.5690E-18	7.1041E+06	2.6048E+01
Xe-135	1.4044E-09	5.4995E-19	2.4532E+06	5.1963E+01
Cs-135	3.3126E-18	2.8752E-18	1.2826E+07	1.2257E-07

Control Room Transport Group Inventory:

Time (h) = 24.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.0110E+10	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.1895E+08	0.0000E+00	0.0000E+00
Organic I (atoms)	1.7869E+05	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 24.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 24.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

trol Room Exhaust Transport Group Inventory:

	Pathway Filter
e (h) = 24.0000	
le gases (atoms)	0.0000E+00
mental I (atoms)	0.0000E+00
anic I (atoms)	0.0000E+00
osols (kg)	0.0000E+00

lusion Area Boundary Doses:

e (h) = 96.0000	Whole Body	Thyroid	TEDE
ta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
umulated dose (rem)	4.2594E-01	0.0000E+00	1.4424E+00

er Boundary of the LPZ Doses:

e (h) = 96.0000	Whole Body	Thyroid	TEDE
ta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
umulated dose (rem)	4.7547E-02	0.0000E+00	1.6101E-01

trol Room Doses:

e (h) = 96.0000	Whole Body	Thyroid	TEDE
ta dose (rem)	3.6064E-12	0.0000E+00	3.3485E-10
umulated dose (rem)	2.4782E-02	0.0000E+00	1.0261E+00

trol Room Compartment Nuclide Inventory:

e (h) = 96.0000	Ci	kg	Atoms	Bq

trol Room Transport Group Inventory:

e (h) = 96.0000	Atmosphere	Sump	Overlying Pool
le gases (atoms)	2.5382E-18	0.0000E+00	0.0000E+00
nental I (atoms)	2.9862E-20	0.0000E+00	0.0000E+00
anic I (atoms)	4.4860E-23	0.0000E+00	0.0000E+00
osols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

e (h) = 96.0000	Deposition Surfaces	Recirculating Filter
le gases (atoms)	0.0000E+00	0.0000E+00
nental I (atoms)	0.0000E+00	0.0000E+00
anic I (atoms)	0.0000E+00	0.0000E+00
osols (kg)	0.0000E+00	0.0000E+00

iltered Environment to CR Transport Group Inventory:

e (h) = 96.0000	Pathway Filter
le gases (atoms)	0.0000E+00
nental I (atoms)	0.0000E+00
anic I (atoms)	0.0000E+00
osols (kg)	0.0000E+00

trol Room Exhaust Transport Group Inventory:

e (h) = 96.0000	Pathway Filter
le gases (atoms)	0.0000E+00
nental I (atoms)	0.0000E+00
anic I (atoms)	0.0000E+00

Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 720.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2594E-01	0.0000E+00	1.4424E+00

Outer Boundary of the LPZ Doses:

Time (h) = 720.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7547E-02	0.0000E+00	1.6101E-01

Control Room Doses:

Time (h) = 720.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	3.0169E-40	0.0000E+00	4.1186E-38
Accumulated dose (rem)	2.4782E-02	0.0000E+00	1.0261E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 720.0000	Ci	kg	Atoms	Bq

Control Room Transport Group Inventory:

Time (h) = 720.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5937-257	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.8750-259	0.0000E+00	0.0000E+00
Organic I (atoms)	2.8167-262	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 720.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 720.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 720.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

2223

##
I-131 Summary
##

Time (hr)	Containment I-131 (Curies)	Environment I-131 (Curies)	Control Room I-131 (Curies)
0.000	4.3771E-01	1.8194E+02	1.9834E-01
0.275	0.0000E+00	1.8242E+02	1.5580E-01
0.525	0.0000E+00	1.8245E+02	1.2486E-01
0.775	0.0000E+00	1.8248E+02	1.0006E-01
1.025	0.0000E+00	1.8250E+02	8.0194E-02
1.275	0.0000E+00	1.8251E+02	6.4269E-02
1.525	0.0000E+00	1.8253E+02	5.1507E-02
1.775	0.0000E+00	1.8254E+02	4.1278E-02
2.000	0.0000E+00	1.8254E+02	3.3822E-02
2.250	0.0000E+00	1.8255E+02	2.7100E-02
2.500	0.0000E+00	1.8256E+02	2.1715E-02
2.750	0.0000E+00	1.8256E+02	1.7399E-02
3.000	0.0000E+00	1.8256E+02	1.3941E-02
3.250	0.0000E+00	1.8257E+02	1.1171E-02
3.500	0.0000E+00	1.8257E+02	8.9506E-03
3.750	0.0000E+00	1.8257E+02	7.1718E-03
4.000	0.0000E+00	1.8257E+02	5.7465E-03
4.250	0.0000E+00	1.8257E+02	4.6045E-03
4.500	0.0000E+00	1.8257E+02	3.6894E-03
4.750	0.0000E+00	1.8258E+02	2.9562E-03
5.000	0.0000E+00	1.8258E+02	2.3687E-03
5.250	0.0000E+00	1.8258E+02	1.8979E-03
5.500	0.0000E+00	1.8258E+02	1.5208E-03
5.750	0.0000E+00	1.8258E+02	1.2185E-03
6.000	0.0000E+00	1.8258E+02	9.7636E-04
6.250	0.0000E+00	1.8258E+02	7.8232E-04
6.500	0.0000E+00	1.8258E+02	6.2685E-04
6.750	0.0000E+00	1.8258E+02	5.0227E-04
7.000	0.0000E+00	1.8258E+02	4.0245E-04
7.250	0.0000E+00	1.8258E+02	3.2247E-04
7.500	0.0000E+00	1.8258E+02	2.5838E-04
7.750	0.0000E+00	1.8258E+02	2.0703E-04
8.000	0.0000E+00	1.8258E+02	1.6589E-04
8.400	0.0000E+00	1.8258E+02	1.1637E-04
8.700	0.0000E+00	1.8258E+02	8.9205E-05
9.000	0.0000E+00	1.8258E+02	6.8379E-05
9.300	0.0000E+00	1.8258E+02	5.2415E-05
9.600	0.0000E+00	1.8258E+02	4.0178E-05
9.900	0.0000E+00	1.8258E+02	3.0797E-05
10.200	0.0000E+00	1.8258E+02	2.3607E-05
24.000	0.0000E+00	1.8258E+02	1.1520E-10
96.000	0.0000E+00	1.8258E+02	2.2331E-38
720.000	0.0000E+00	1.8258E+02	1.4904-278

##
Cumulative Dose Summary
##

Time (hr)	Exclusion Area Bounda Thyroid (rem)	TEDE (rem)	Outer Boundary of the Thyroid (rem)	TEDE (rem)	Control Room Thyroid (rem)	TEDE (rem)
0.000	0.0000E+00	1.4377E+00	0.0000E+00	1.6048E-01	0.0000E+00	4.5686E-05
0.275	0.0000E+00	1.4415E+00	0.0000E+00	1.6091E-01	0.0000E+00	2.2323E-01
0.525	0.0000E+00	1.4417E+00	0.0000E+00	1.6094E-01	0.0000E+00	3.8370E-01
0.775	0.0000E+00	1.4419E+00	0.0000E+00	1.6096E-01	0.0000E+00	5.1208E-01
1.025	0.0000E+00	1.4421E+00	0.0000E+00	1.6097E-01	0.0000E+00	6.1479E-01
1.275	0.0000E+00	1.4422E+00	0.0000E+00	1.6099E-01	0.0000E+00	6.9696E-01

1.525	0.0000E+00	1.4423E+00	0.0000E+00	1.6100E-01	0.0000E+00	7.6271E-01
1.775	0.0000E+00	1.4423E+00	0.0000E+00	1.6101E-01	0.0000E+00	8.1533E-01
2.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	8.5363E-01
2.250	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	8.8808E-01
2.500	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	9.1564E-01
2.750	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	9.3770E-01
3.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	9.5535E-01
3.250	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	9.6947E-01
3.500	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	9.8077E-01
3.750	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	9.8982E-01
4.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	9.9706E-01
4.250	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0028E+00
4.500	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0075E+00
4.750	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0112E+00
5.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0142E+00
5.250	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0165E+00
5.500	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0184E+00
5.750	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0200E+00
6.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0212E+00
6.250	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0222E+00
6.500	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0229E+00
6.750	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0236E+00
7.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0241E+00
7.250	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0245E+00
7.500	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0248E+00
7.750	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0250E+00
8.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0253E+00
8.400	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0255E+00
8.700	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0256E+00
9.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0257E+00
9.300	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0258E+00
9.600	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0259E+00
9.900	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0259E+00
10.200	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0260E+00
24.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0261E+00
96.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0261E+00
720.000	0.0000E+00	1.4424E+00	0.0000E+00	1.6101E-01	0.0000E+00	1.0261E+00

```
#################################################################
                    Worst Two-Hour Doses
   Note: All of the dose locations are shown below but the
         worst two-hour dose is only meaningful for the EAB
         dose location.  Please disregard the two-hour worst
         doses for the other dose locations
#################################################################
```

Exclusion Area Boundary

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
0.0	1.3779E-03	0.0000E+00	4.7275E-03

Outer Boundary of the LPZ

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
0.0	1.5382E-04	0.0000E+00	5.2772E-04

Control Room

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
0.0	2.1084E-02	0.0000E+00	8.5359E-01

```
 ##########################################################################
          RADTRAD Version 3.02 run on 11/13/2001  at 13:54:31
 ##########################################################################

 ##########################################################################
                        File information
 ##########################################################################

 Plant file name            = pnpp_fha.psf
 Inventory file name        = d:\hwagage\computer codes\radtrad\run
batch\perry\pnpp_fha.nif
 Scenario file name         = NEW_SDF.SDF
 Release file name          = d:\hwagage\computer codes\radtrad\run
batch\perry\pnpp_fha.rft
 Dose conversion file name = d:\hwagage\computer codes\radtrad\run
batch\perry\pnpp_fha.dcf


     #####    ####   #####    #  #     #  #####   #    #  #####
     #    #  #       #        #  ##    #  #    #  #    #    #
     #    #  #       #        #  # #   #  #    #  #    #    #
     #####    ####   ####     #  #  #  #  #####   #    #    #
     #            #  #        #  #   # #  #       #    #    #
     #            #  #        #  #    ##  #       #    #    #
     #        ####   #        #  #     #  #        ####     #


 Radtrad 3.02 1/5/2000
 perry fha: sensitivity case 1
 Nuclide Inventory File:
 d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.nif
 Plant Power Level:
  3.8332E+03
 Compartments:
   3
 Compartment 1:
 Containment
   3
  1.0000E+00
   0
   0
   0
   0
   0
 Compartment 2:
 Environment
   2
  0.0000E+00
   0
   0
   0
   0
   0
 Compartment 3:
 Control Room
   1
  3.6707E+05
   0
   0
   0
   0
   0
 Pathways:
   3
 Pathway 1:
 Unfiltered Release to Environment
   1
   2
   4
 Pathway 2:
 Unfiltered Environment to CR
   2
   3
   2
 Pathway 3:
 Control Room Exhaust
   3
   2
   2
 End of Plant Model File
 Scenario Description Name:

 Plant Model Filename:

 Source Term:
   1
   1   1.0000E+00
 d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.dcf
 d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.rft
   0.0000E+00
   1
   0.0000E+00   0.9985E+00   0.0015E+00   1.0000E+00
 Overlying Pool:
   0
   0.0000E+00
   0
   0
   0
   0
 Compartments:
   3
 Compartment 1:
   0
   1
   0
   0
   0
   0
   0
   0
   0
 Compartment 2:
   0
   1
   0
   0
   0
   0
   0
   0
   0
 Compartment 3:
```

```
  1
  1
  0
  0
  0
  0
  0
  0
  0
Pathways·
  3
Pathway 1.
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  2
  0.0000E+00   1.0000E+10
  7.2000E+02   0.0000E+00
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0.0001E+00   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 3:
  0
  0
  0
  0
  0
  1
  4
  0.0000E+00   6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0.0001E+00   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
Exclusion Area Boundary
  2
  1
  2
  0.0000E+00   4.3000E-04
  2.0000E+00   0.0000E+00
  1
  3
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  0
Location 2:
Outer Boundary of the LPZ
  2
  1
  2
  0.0000E+00   4.8000E-05
  2.0000E+00   0.0000E+00
  1
  3
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  0
Location 3:
Control Room
  3
  0
  1
  2
  0.0000E+00   3.5000E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Effective Volume Location:
  1
  2
  0.0000E+00   3.5000E-04
  2.0000E+00   0.0000E+00
Simulation Parameters:
  4
  0.0000E+00   2.5000E-02
  8.0000E+00   1.0000E-01
  2.4000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Output Filename:

  1
  1
  1
  0
  0
End of Scenario File
```

```
##########################################################################
          RADTRAD Version 3.02 run on 11/13/2001  at 13:54:31
##########################################################################

##########################################################################
                    Plant Description
##########################################################################

Number of Nuclides =  23

Inventory Power =   3.8332E+03 MWth
Plant Power Level =   3.8332E+03 MWth

Number of compartments    =   3

Compartment information

Compartment number  1  (Source term fraction =   1.0000E+00
)
Name: Containment
Compartment volume =   1.0000E+00 (Cubic feet)
Pathways into and out of compartment  1
    Pathway to compartment number   2: Unfiltered Release to Environment

Compartment number  2
Name: Environment
Pathways into and out of compartment  2
    Pathway to compartment number   3: Unfiltered Environment to CR
    Pathway from compartment number 1: Unfiltered Release to Environment
    Pathway from compartment number 3: Control Room Exhaust

Compartment number  3
Name: Control Room
Compartment volume =   3.6707E+05 (Cubic feet)
Pathways into and out of compartment  3
    Pathway to compartment number   2: Control Room Exhaust
    Pathway from compartment number 2: Unfiltered Environment to CR

Total number of pathways =   3
```

```
#######################################################################
          RADTRAD Version 3.02 run on 11/13/2001  at 13:54:31
#######################################################################

#######################################################################
                      Scenario Description
#######################################################################

Radioactive Decay is enabled
Calculation of Daughters is enabled
RELEASE_NAME = Perry FHA
Release Fractions and Timings
                GAP        EARLY IN-VESSEL
              0.0001 hrs     0.0000 hrs
NOBLES        1.0000E+00     0.0000E+00
IODINE        1.0000E+00     0.0000E+00
CESIUM        1.0000E+00     0.0000E+00
TELLURIUM     0.0000E+00     0.0000E+00
STRONTIUM     0.0000E+00     0.0000E+00
BARIUM        0.0000E+00     0.0000E+00
RUTHENIUM     0.0000E+00     0.0000E+00
CERIUM        0.0000E+00     0.0000E+00
LANTHANUM     0.0000E+00     0.0000E+00

Iodine fractions
     Aerosol     =   0.0000E+00
     Elemental   =   9.9850E-01
     Organic     =   1.5000E-03

COMPARTMENT DATA

Compartment number   1: Containment

Compartment number   2: Environment

Compartment number   3: Control Room

PATHWAY DATA

Pathway number  1: Unfiltered Release to Environment

     Convection Data
      Time (hr)          Flow Rate (% / day)
       0.0000E+00              1.0000E+10
       7.2000E+02              0.0000E+00

Pathway number  2: Unfiltered Environment to CR

     Pathway Filter: Removal Data

       Time (hr)      Flow Rate          Filter Efficiencies (%)
                       (cfm)        Aerosol      Elemental     Organic
       0.0000E+00    6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
       1.0000E-04    0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
       2.0000E+00    5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
       7.2000E+02    5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00

Pathway number  3: Control Room Exhaust

     Pathway Filter: Removal Data

       Time (hr)      Flow Rate          Filter Efficiencies (%)
                       (cfm)        Aerosol      Elemental     Organic
       0.0000E+00    6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
       1.0000E-04    0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
       2.0000E+00    5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00
       7.2000E+02    5.4000E+03   0.0000E+00   0.0000E+00   0.0000E+00

LOCATION DATA
     Location Exclusion Area Boundary     is in compartment   2

     Location X/Q Data
      Time (hr)          X/Q (s * m^-3)
       0.0000E+00          4.3000E-04
       2.0000E+00          0.0000E+00

     Location Breathing Rate Data
      Time (hr)          Breathing Rate (m^3 * sec^-1)
       0.0000E+00                 3.5000E-04
       8.0000E+00                 1.8000E-04
       2.4000E+01                 2.3000E-04
     Location Outer Boundary of the LPZ     is in compartment   2

     Location X/Q Data
      Time (hr)          X/Q (s * m^-3)
       0.0000E+00          4.8000E-05
       2.0000E+00          0.0000E+00

     Location Breathing Rate Data
      Time (hr)          Breathing Rate (m^3 * sec^-1)
       0.0000E+00                 3.5000E-04
       8.0000E+00                 1.8000E-04
       2.4000E+01                 2.3000E-04
     Location Control Room     is in compartment   3

     Location X/Q Data
      Time (hr)          X/Q (s * m^-3)
       0.0000E+00          3.5000E-04
       2.0000E+00          0.0000E+00

     Location Breathing Rate Data
      Time (hr)          Breathing Rate (m^3 * sec^-1)
       0.0000E+00                 3.5000E-04
       7.2000E+02                 0.0000E+00

     Location Occupancy Factor Data
      Time (hr)          Occupancy Factor
       0.0000E+00            1.0000E+00
       2.4000E+01            6.0000E-01
       9.6000E+01            4.0000E-01
       7.2000E+02            0.0000E+00

USER SPECIFIED TIME STEP DATA - SUPPLEMENTAL TIME STEPS
     Time              Time step
      0.0000E+00        2.5000E-02
      8.0000E+00        1.0000E-01
      2.4000E+01        4.0000E-01
      7.2000E+02        0.0000E+00
```

```
##########################################################################
          RADTRAD Version 3.02 run on 11/13/2001  at 13:54:31
##########################################################################

 ####   #    #  #####  #####   #    #  #####
#    #  #    #    #    #    #  #    #    #
#    #  #    #    #    #    #  #    #    #
#    #  #    #    #    #####   #    #    #
#    #  #    #    #    #       #    #    #
#    #  #    #    #    #       #    #    #
 ####    ####     #    #        ####     #

##########################################################################
          Dose, Detailed model and Detailed Inventory Output
##########################################################################
```

Exclusion Area Boundary Doses:

Time (h) = 0.0001	Whole Body	Thyroid	TEDE
Delta dose (rem)	4.2456E-01	0.0000E+00	1.4377E+00
Accumulated dose (rem)	4.2456E-01	0.0000E+00	1.4377E+00

Outer Boundary of the LPZ Doses:

Time (h) = 0.0001	Whole Body	Thyroid	TEDE
Delta dose (rem)	4.7393E-02	0.0000E+00	1.6048E-01
Accumulated dose (rem)	4.7393E-02	0.0000E+00	1.6048E-01

Control Room Doses:

Time (h) = 0.0001	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.2078E-06	0.0000E+00	4.5686E-05
Accumulated dose (rem)	1.2078E-06	0.0000E+00	4.5686E-05

Control Room Compartment Nuclide Inventory:

Time (h) = 0.0001	Ci	kg	Atoms	Bq
Br-82	6.0733E-04	5.6097E-13	4.1198E+12	2.2471E+07
Br-83	1.6156E-05	1.0227E-15	7.4200E+09	5.9776E+05
Kr-83m	1.2500E-02	6.0583E-13	4.3957E+12	4.6248E+08
Kr-85	8.1467E-01	2.0751E-06	1.4702E+19	3.0143E+10
Kr-85m	1.6234E-01	1.9727E-11	1.3976E+14	6.0067E+09
Kr-87	2.6301E-05	9.2854E-16	6.4273E+09	9.7315E+05
Kr-88	5.0143E-02	3.9989E-12	2.7366E+13	1.8553E+09
Rb-87	4.4281E-24	5.0613E-20	3.5034E+05	1.6384E-13
Rb-88	1.1716E-05	9.7601E-17	6.6792E+08	4.3349E+05
I-129	6.8183E-09	3.8601E-08	1.8020E+17	2.5228E+02
I-130	1.4897E-03	7.6381E-13	3.5383E+12	5.5118E+07
I-131	1.9834E-01	1.5999E-09	7.3546E+15	7.3387E+09
I-132	1.5753E-01	1.5262E-11	6.9627E+13	5.8288E+09
I-133	1.2391E-01	1.0938E-10	4.9529E+14	4.5847E+09
I-134	6.5286E-09	2.4473E-19	1.0999E+06	2.4156E+02
I-135	2.0391E-02	5.8063E-12	2.5901E+13	7.5446E+08
Xe-129m	2.2597E-04	1.7859E-12	8.3372E+12	8.3607E+06
Xe-131m	2.9842E-01	3.5627E-09	1.6378E+16	1.1041E+10
Xe-133	5.0066E+01	2.6747E-07	1.2111E+18	1.8525E+12
Xe-133m	1.5255E+00	3.3997E-09	1.5394E+16	5.6442E+10
Xe-135	1.3783E+01	5.3973E-09	2.4076E+16	5.0998E+11
Xe-135m	6.5313E-01	7.1700E-12	3.1984E+13	2.4166E+10
Cs-135	4.7418E-14	4.1157E-14	1.8359E+11	1.7545E-03

Control Room Transport Group Inventory:

Time (h) = 0.0001	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5969E+19	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.8787E+17	0.0000E+00	0.0000E+00
Organic I (atoms)	2.8223E+14	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 0.0001	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 0.0001	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 0.0001	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 2.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.0214E-03	0.0000E+00	3.4587E-03
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 2.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.1402E-04	0.0000E+00	3.8609E-04
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 2.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	4.3741E-02	0.0000E+00	1.8080E+00
Accumulated dose (rem)	4.3742E-02	0.0000E+00	1.8080E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 2.0000	Ci	kg	Atoms	Bq
Br-82	5.8394E-04	5.3937E-13	3.9611E+12	2.1606E+07
Br-83	9.0455E-06	5.7258E-16	4.1544E+09	3.3468E+05
Kr-83m	5.8665E-03	2.8434E-13	2.0631E+12	2.1706E+08
Kr-85	8.1466E-01	2.0750E-06	1.4701E+19	3.0143E+10

Kr-85m	1.1914E-01	1.4477E-11	1.0257E+14	4.4081E+09
Kr-87	8.8422E-06	3.1216E-16	2.1608E+09	3.2716E+05
Kr-88	3.0777E-02	2.4545E-12	1.6797E+13	1.1388E+09
Rb-87	5.3930E-20	6.1643E-16	4.2669E+09	1.9954E-09
Rb-88	3.4843E-02	2.9027E-13	1.9864E+12	1.2892E+09
I-129	6.8183E-09	3.8601E-08	1.8020E+17	2.5228E+02
I-130	1.3316E-03	6.8277E-13	3.1629E+12	4.9271E+07
I-131	1.9692E-01	1.5884E-09	7.3020E+15	7.2861E+09
I-132	8.6223E-02	8.3532E-12	3.8109E+13	3.1902E+09
I-133	1.1592E-01	1.0233E-10	4.6335E+14	4.2892E+09
I-134	1.3431E-09	5.0346E-20	2.2626E+05	4.9693E+01
I-135	1.6533E-02	4.7078E-12	2.1001E+13	6.1173E+08
Xe-129m	2.2434E-04	1.7731E-12	8.2773E+12	8.3006E+06
Xe-131m	2.9698E-01	3.5456E-09	1.6299E+16	1 0988E+10
Xe-133	4.9536E+01	2.6464E-07	1.1983E+18	1.8328E+12
Xe-133m	1.4858E+00	3.3113E-09	1.4993E+16	5.4975E+10
Xe-135	1.1852E+01	4.6412E-09	2.0704E+16	4.3854E+11
Xe-135m	5.5597E-03	6.1033E-14	2.7226E+11	2.0571E+08
Cs-135	8.8064E-10	7.6435E-10	3.4096E+15	3.2584E+01

Control Room Transport Group Inventory:

Time (h) = 2.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5969E+19	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.8787E+17	0.0000E+00	0.0000E+00
Organic I (atoms)	2.8223E+14	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 2.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 2.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 2.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 8.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 8.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 8.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	2.1440E-02	0.0000E+00	9.9924E-01
Accumulated dose (rem)	6.5182E-02	0.0000E+00	2.8073E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 8.0000	Ci	kg	Atoms	Bq
Br-82	2.6012E-06	2.4027E-15	1.7646E+10	9.6246E+04
Br-83	7.9559E-09	5.0361E-19	3.6540E+06	2.9437E+02
Kr-83m	3.0436E-06	1.4752E-16	1.0703E+09	1.1261E+05
Kr-85	4.0826E-03	1.0399E-08	7.3674E+16	1.5106E+08
Kr-85m	2.3598E-04	2.8674E-14	2.0315E+11	8.7311E+06
Kr-87	1.6835E-09	5.9434E-20	4.1140E+05	6.2290E+01
Kr-88	3.5664E-05	2.8442E-15	1.9464E+10	1.3196E+06
Rb-87	4.0195E-22	4.5943E-18	3.1802E+07	1.4872E-11
Rb-88	4.1000E-05	3.4156E-16	2.3374E+09	1.5170E+06
I-129	3.4171E-11	1.9345E-10	9.0311E+14	1.2643E+00
I-130	4.7669E-06	2.4441E-15	1.1322E+10	1.7637E+05
I-131	9.6586E-04	7.7908E-12	3.5815E+13	3.5737E+07
I-132	7.0844E-05	6.8633E-15	3.1312E+10	2.6212E+06
I-133	4.7568E-04	4.1991E-13	1.9013E+12	1.7600E+07
I-135	4.4166E-05	1.2576E-14	5.6101E+10	1.6341E+06
Xe-129m	1.1002E-06	8.6956E-15	4.0594E+10	4.0708E+04
Xe-131m	1.4670E-03	1.7514E-11	8.0514E+13	5.4279E+07
Xe-133	2.4044E-01	1.2845E-09	5.8162E+15	8.8961E+09
Xe-133m	6.8805E-03	1.5334E-11	6.9432E+13	2.5458E+08
Xe-135	3.7613E-02	1.4729E-11	6.5703E+13	1.3917E+09
Xe-135m	7.3176E-06	8.0332E-17	3.5835E+08	2.7075E+05
Cs-135	1.4255E-11	1.2373E-11	5.5194E+13	5.2745E-01

Control Room Transport Group Inventory:

Time (h) = 8.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	8.0029E+16	0.0000E+00	0.0000E+00
Elemental I (atoms)	9.4155E+14	0.0000E+00	0.0000E+00
Organic I (atoms)	1.4145E+12	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 8.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 8.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Pathway

Time (h) = 8.0000	Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 24.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 24.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 24.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	7.9420E-05	0.0000E+00	4.8094E-03
Accumulated dose (rem)	6.5262E-02	0.0000E+00	2.8121E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 24 0000	Ci	kg	Atoms	Bq
Kr-85	3.0026E-09	7.6480E-15	5.4185E+10	1.1110E+02
I-129	2.5135E-17	1.4230E-16	6.6428E+08	9.2998E-07
I-131	6.7076E-10	5.4104E-18	2.4872E+07	2.4818E+01
I-133	2.0529E-10	1.8122E-19	8.2055E+05	7.5957E+00
Xe-131m	1.0383E-09	1.2395E-17	5.6982E+07	3.8415E+01
Xe-133	1.6235E-07	8.6731E-16	3.9271E+09	6.0068E+03
Xe-133m	4.0989E-09	9.1350E-18	4.1363E+07	1.5166E+02
Xe-135	8.1770E-09	3.2020E-18	1.4284E+07	3.0255E+02
Cs-135	1.9287E-17	1.6740E-17	7.4676E+07	7.1363E-07

Control Room Transport Group Inventory:

Time (h) = 24.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	5.8866E+10	0.0000E+00	0.0000E+00
Elemental I (atoms)	6.9256E+08	0.0000E+00	0.0000E+00
Organic I (atoms)	1.0404E+06	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 24.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 24.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 24.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 96.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 96.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 96.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	2.0998E-11	0.0000E+00	1.9496E-09
Accumulated dose (rem)	6.5262E-02	0.0000E+00	2.8121E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 96.0000	Ci	kg	Atoms	Bq

Control Room Transport Group Inventory:

Time (h) = 96.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.4778E-17	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.7387E-19	0.0000E+00	0.0000E+00
Organic I (atoms)	2.6119E-22	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 96.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 96.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 96.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 720.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 720 0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 720.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.7565E-39	0.0000E+00	2.3980E-37
Accumulated dose (rem)	6.5262E-02	0.0000E+00	2.8121E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 720.0000	Ci	kg	Atoms	Bq

Control Room Transport Group Inventory:

Time (h) = 720.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	9.2790-257	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.0917-258	0.0000E+00	0.0000E+00
Organic I (atoms)	1.6400-261	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 720.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 720.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 720.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

2223

##
I-131 Summary
##

Time (hr)	Containment I-131 (Curies)	Environment I-131 (Curies)	Control Room I-131 (Curies)
0.000	4.3771E-01	1.8194E+02	1.9834E-01
0.275	0.0000E+00	1.8238E+02	1.9815E-01
0.525	0.0000E+00	1.8238E+02	1.9797E-01
0.775	0.0000E+00	1.8238E+02	1.9779E-01
1.025	0.0000E+00	1.8238E+02	1.9761E-01
1.275	0.0000E+00	1.8238E+02	1.9744E-01
1.525	0.0000E+00	1.8238E+02	1.9726E-01
1.775	0.0000E+00	1.8238E+02	1.9708E-01
2.000	0.0000E+00	1.8238E+02	1.9692E-01
2.250	0.0000E+00	1.8242E+02	1.5779E-01
2.500	0.0000E+00	1.8245E+02	1.2643E-01
2.750	0.0000E+00	1.8248E+02	1.0130E-01
3.000	0.0000E+00	1.8250E+02	8.1171E-02
3.250	0.0000E+00	1.8251E+02	6.5039E-02
3.500	0.0000E+00	1.8252E+02	5.2114E-02
3.750	0.0000E+00	1.8253E+02	4.1757E-02
4.000	0.0000E+00	1.8254E+02	3.3458E-02
4.250	0.0000E+00	1.8255E+02	2.6809E-02
4.500	0.0000E+00	1.8255E+02	2.1481E-02
4.750	0.0000E+00	1.8256E+02	1.7212E-02
5.000	0.0000E+00	1.8256E+02	1.3791E-02
5.250	0.0000E+00	1.8257E+02	1.1050E-02
5.500	0.0000E+00	1.8257E+02	8.8544E-03
5.750	0.0000E+00	1.8257E+02	7.0947E-03
6.000	0.0000E+00	1.8257E+02	5.6847E-03
6.250	0.0000E+00	1.8257E+02	4.5550E-03
6.500	0.0000E+00	1.8257E+02	3.6497E-03
6.750	0.0000E+00	1.8257E+02	2.9244E-03
7.000	0.0000E+00	1.8257E+02	2.3432E-03
7.250	0.0000E+00	1.8257E+02	1.8775E-03
7.500	0.0000E+00	1.8257E+02	1.5044E-03
7.750	0.0000E+00	1.8257E+02	1.2054E-03
8.000	0.0000E+00	1.8258E+02	9.6586E-04
8.400	0.0000E+00	1.8258E+02	6.7757E-04
8.700	0.0000E+00	1.8258E+02	5.1938E-04
9.000	0.0000E+00	1.8258E+02	3.9812E-04
9.300	0.0000E+00	1.8258E+02	3.0518E-04
9.600	0.0000E+00	1.8258E+02	2.3393E-04
9.900	0.0000E+00	1.8258E+02	1.7931E-04
10.200	0.0000E+00	1.8258E+02	1.3745E-04
24.000	0.0000E+00	1.8258E+02	6.7076E-10
96.000	0.0000E+00	1.8258E+02	1.3002E-37
720.000	0.0000E+00	1.8258E+02	8.6776-278

##
Cumulative Dose Summary
##

Time (hr)	Exclusion Area Bounda Thyroid (rem)	TEDE (rem)	Outer Boundary of the Thyroid (rem)	TEDE (rem)	Control Room Thyroid (rem)	TEDE (rem)
0.000	0.0000E+00	1.4377E+00	0.0000E+00	1.6048E-01	0.0000E+00	4.5686E-05
0.275	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.5085E-01
0.525	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	4.7825E-01
0.775	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	7.0504E-01
1.025	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	9.3124E-01
1.275	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	1.1569E+00
1.525	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	1.3819E+00
1.775	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	1.6064E+00
2.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	1.8080E+00
2.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.0086E+00

2.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.1691E+00
2.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2975E+00
3.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.4003E+00
3.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.4825E+00
3.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.5483E+00
3.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.6010E+00
4.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.6431E+00
4.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.6768E+00
4.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7038E+00
4.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7254E+00
5.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7427E+00
5.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7566E+00
5.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7676E+00
5.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7765E+00
6.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7836E+00
6.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7893E+00
6.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7938E+00
6.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.7975E+00
7.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8004E+00
7.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8027E+00
7.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8046E+00
7.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8061E+00
8.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8073E+00
8.400	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8087E+00
8.700	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8095E+00
9.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8101E+00
9.300	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8106E+00
9.600	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8109E+00
9.900	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8112E+00
10.200	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8114E+00
24.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8121E+00
96.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8121E+00
720.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.8121E+00

```
#######################################################################
                    Worst Two-Hour Doses
   Note: All of the dose locations are shown below but the
         worst two-hour dose is only meaningful for the EAB
         dose location.  Please disregard the two-hour worst
         doses for the other dose locations
#######################################################################
```

Exclusion Area Boundary

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
0.0	1.0214E-03	0.0000E+00	3.4587E-03

Outer Boundary of the LPZ

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
0.0	1.1402E-04	0.0000E+00	3.8609E-04

Control Room

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
0.0	4.3741E-02	0.0000E+00	1.8080E+00

```
#######################################################################
          RADTRAD Version 3.02 run on 11/13/2001  at 13:56:28
#######################################################################

#######################################################################
                      File information
#######################################################################

 Plant file name            = pnpp_fha.psf
 Inventory file name        = d:\hwagage\computer codes\radtrad\run
batch\perry\pnpp_fha.nif
 Scenario file name         = NEW_SDF.SDF
 Release file name          = d:\hwagage\computer codes\radtrad\run
batch\perry\pnpp_fha.rft
 Dose conversion file name  = d:\hwagage\computer codes\radtrad\run
batch\perry\pnpp_fha.dcf


     #####    ####   #####    #  #     #  #####   #    #  #####
     #    #  #       #        #  ##    #  #    #  #    #    #
     #    #  #       #        #  # #   #  #    #  #    #    #
     #####    ####   ####     #  #  #  #  #####   #    #    #
     #            #  #        #  #   # #  #       #    #    #
     #            #  #        #  #    ##  #       #    #    #
     #        ####   #        #  #     #  #        ####     #


 Radtrad 3.02 1/5/2000
 perry fha: sensitivity case 2
 Nuclide Inventory File:
 d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.nif
 Plant Power Level:
  3.8332E+03
 Compartments:
   3
 Compartment 1:
 Containment
   3
  1.0000E+00
   0
   0
   0
   0
   0
 Compartment 2:
 Environment
   2
  0.0000E+00
   0
   0
   0
   0
   0
 Compartment 3:
 Control Room
   1
  3.6707E+05
   0
   0
   1
   0
   0
 Pathways:
   3
 Pathway 1:
 Unfiltered Release to Environment
   1
   2
   4
 Pathway 2:
 Unfiltered Environment to CR
   2
   3
   2
 Pathway 3:
 Control Room Exhaust
   3
   2
   2
 End of Plant Model File
 Scenario Description Name:

 Plant Model Filename:

 Source Term:
   1
   1   1.0000E+00
 d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.dcf
 d:\hwagage\computer codes\radtrad\run batch\perry\pnpp_fha.rft
   0.0000E+00
   1
   0.0000E+00   0.9985E+00   0.0015E+00   1.0000E+00
 Overlying Pool:
   0
   0.0000E+00
   0
   0
   0
   0
 Compartments:
   3
 Compartment 1:
   0
   1
   0
   0
   0
   0
   0
   0
   0
 Compartment 2:
   0
   1
   0
   0
   0
   0
   0
   0
   0
 Compartment 3:
   1
```

```
  1
  0
  0
  0
  0
  1
  2.7000E+04
  3
  0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  2.0000E+00   5.0000E+01   5.0000E+01   5.0000E+01
  7.2000E+02   5.0000E+01   5.0000E+01   5.0000E+01
  0
  0
Pathways:
  3
Pathway 1:
  0
  0
  0
  0
  0
  0
  0
  0
  0
  0
  1
  2
  0.0000E+00   1.0000E+10
  7.2000E+02   0.0000E+00
  0
Pathway 2:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0.0001E+00   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Pathway 3:
  0
  0
  0
  0
  0
  1
  3
  0.0000E+00   6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
  0.0001E+00   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  7.2000E+02   0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
  0
  0
  0
  0
  0
  0
Dose Locations:
  3
Location 1:
Exclusion Area Boundary
  2
  1
  2
  0.0000E+00   4.3000E-04
  2.0000E+00   0.0000E+00
  1
  3
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  0
Location 2:
Outer Boundary of the LPZ
  2
  1
  2
  0.0000E+00   4.8000E-05
  2.0000E+00   0.0000E+00
  1
  3
  0.0000E+00   3.5000E-04
  8.0000E+00   1.8000E-04
  2.4000E+01   2.3000E-04
  0
Location 3:
Control Room
  3
  0
  1
  2
  0.0000E+00   3.5000E-04
  7.2000E+02   0.0000E+00
  1
  4
  0.0000E+00   1.0000E+00
  2.4000E+01   6.0000E-01
  9.6000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Effective Volume Location:
  1
  2
  0.0000E+00   3.5000E-04
  2.0000E+00   0.0000E+00
Simulation Parameters:
  4
  0.0000E+00   2.5000E-02
  8.0000E+00   1.0000E-01
  2.4000E+01   4.0000E-01
  7.2000E+02   0.0000E+00
Output Filename:

  1
  1
  1
  0
  0
End of Scenario File
```

```
##########################################################################
          RADTRAD Version 3.02 run on 11/13/2001  at 13:56:28
##########################################################################

##########################################################################
                      Plant Description
##########################################################################

Number of Nuclides =  23

Inventory Power =   3.8332E+03 MWth
Plant Power Level =   3.8332E+03 MWth

Number of compartments    =   3

Compartment information

Compartment number  1  (Source term fraction =   1.0000E+00
)
Name: Containment
Compartment volume =   1.0000E+00 (Cubic feet)
Pathways into and out of compartment  1
    Pathway to compartment number   2: Unfiltered Release to Environment

Compartment number  2
Name: Environment
Pathways into and out of compartment  2
    Pathway to compartment number   3: Unfiltered Environment to CR
    Pathway from compartment number 1: Unfiltered Release to Environment
    Pathway from compartment number 3: Control Room Exhaust

Compartment number  3
Name: Control Room
Compartment volume =   3.6707E+05 (Cubic feet)
Removal devices within compartment:
     Filter(s)
Pathways into and out of compartment  3
    Pathway to compartment number   2: Control Room Exhaust
    Pathway from compartment number 2: Unfiltered Environment to CR

Total number of pathways =   3
```

```
#######################################################################
          RADTRAD Version 3.02 run on 11/13/2001  at 13:56:28
#######################################################################

#######################################################################
                    Scenario Description
#######################################################################

Radioactive Decay is enabled
Calculation of Daughters is enabled
RELEASE_NAME = Perry FHA
Release Fractions and Timings
                GAP        EARLY IN-VESSEL
              0.0001 hrs     0.0000 hrs
NOBLES        1.0000E+00     0.0000E+00
IODINE        1.0000E+00     0.0000E+00
CESIUM        1.0000E+00     0.0000E+00
TELLURIUM     0.0000E+00     0.0000E+00
STRONTIUM     0.0000E+00     0.0000E+00
BARIUM        0.0000E+00     0.0000E+00
RUTHENIUM     0.0000E+00     0.0000E+00
CERIUM        0.0000E+00     0.0000E+00
LANTHANUM     0.0000E+00     0.0000E+00

Iodine fractions
     Aerosol     =  0.0000E+00
     Elemental   =  9.9850E-01
     Organic     =  1.5000E-03

COMPARTMENT DATA

Compartment number   1: Containment

Compartment number   2: Environment

Compartment number   3: Control Room

     Compartment Filter Data

      Time (hr)       Flow Rate          Filter Efficiencies (%)
                       (cfm)        Aerosol     Elemental    Organic
       0.0000E+00    2.7000E+04   0.0000E+00   0.0000E+00   0.0000E+00
       2.0000E+00    2.7000E+04   5.0000E+01   5.0000E+01   5.0000E+01
       7.2000E+02    2.7000E+04   5.0000E+01   5.0000E+01   5.0000E+01

PATHWAY DATA

Pathway number  1: Unfiltered Release to Environment

     Convection Data
      Time (hr)          Flow Rate (% / day)
       0.0000E+00              1.0000E+10
       7.2000E+02              0.0000E+00

Pathway number  2: Unfiltered Environment to CR

     Pathway Filter: Removal Data

      Time (hr)       Flow Rate          Filter Efficiencies (%)
                       (cfm)        Aerosol     Elemental    Organic
       0.0000E+00    6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
       1.0000E-04    0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
       7.2000E+02    0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00

Pathway number  3: Control Room Exhaust

     Pathway Filter: Removal Data

      Time (hr)       Flow Rate          Filter Efficiencies (%)
                       (cfm)        Aerosol     Elemental    Organic
       0.0000E+00    6.6000E+03   0.0000E+00   0.0000E+00   0.0000E+00
       1.0000E-04    0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00
       7.2000E+02    0.0000E+00   0.0000E+00   0.0000E+00   0.0000E+00

LOCATION DATA
     Location Exclusion Area Boundary     is in compartment   2

     Location X/Q Data
      Time (hr)          X/Q (s * m^-3)
       0.0000E+00          4.3000E-04
       2.0000E+00          0.0000E+00

     Location Breathing Rate Data
      Time (hr)          Breathing Rate (m^3 * sec^-1)
       0.0000E+00                 3.5000E-04
       8.0000E+00                 1.8000E-04
       2.4000E+01                 2.3000E-04
     Location Outer Boundary of the LPZ     is in compartment   2

     Location X/Q Data
      Time (hr)          X/Q (s * m^-3)
       0.0000E+00          4.8000E-05
       2.0000E+00          0.0000E+00

     Location Breathing Rate Data
      Time (hr)          Breathing Rate (m^3 * sec^-1)
       0.0000E+00                 3.5000E-04
       8.0000E+00                 1.8000E-04
       2.4000E+01                 2.3000E-04
     Location Control Room     is in compartment   3

     Location X/Q Data
      Time (hr)          X/Q (s * m^-3)
       0.0000E+00          3.5000E-04
       2.0000E+00          0.0000E+00

     Location Breathing Rate Data
      Time (hr)          Breathing Rate (m^3 * sec^-1)
       0.0000E+00                 3.5000E-04
       7.2000E+02                 0.0000E+00

     Location Occupancy Factor Data
      Time (hr)          Occupancy Factor
       0.0000E+00          1.0000E+00
       2.4000E+01          6.0000E-01
       9.6000E+01          4.0000E-01
       7.2000E+02          0.0000E+00

USER SPECIFIED TIME STEP DATA - SUPPLEMENTAL TIME STEPS
     Time            Time step
      0.0000E+00       2.5000E-02
      8.0000E+00       1.0000E-01
      2.4000E+01       4.0000E-01
      7.2000E+02       0.0000E+00
```

```
#######################################################################
          RADTRAD Version 3.02 run on 11/13/2001  at 13:56:28
#######################################################################

 ####   #    #  #####  #####   #    #  #####
#    # #    #    #    #    #  #    #    #
#    # #    #    #    #    #  #    #    #
#    # #    #    #    #####   #    #    #
#    # #    #    #    #       #    #    #
#    # #    #    #    #       #    #    #
 ####    ####     #    #        ####     #

#######################################################################
          Dose, Detailed model and Detailed Inventory Output
#######################################################################
```

Exclusion Area Boundary Doses:

Time (h) = 0.0001	Whole Body	Thyroid	TEDE
Delta dose (rem)	4.2456E-01	0.0000E+00	1.4377E+00
Accumulated dose (rem)	4.2456E-01	0.0000E+00	1.4377E+00

Outer Boundary of the LPZ Doses:

Time (h) = 0.0001	Whole Body	Thyroid	TEDE
Delta dose (rem)	4.7393E-02	0.0000E+00	1.6048E-01
Accumulated dose (rem)	4.7393E-02	0.0000E+00	1.6048E-01

Control Room Doses:

Time (h) = 0.0001	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.2078E-06	0.0000E+00	4.5686E-05
Accumulated dose (rem)	1.2078E-06	0.0000E+00	4.5686E-05

Control Room Compartment Nuclide Inventory:

Time (h) = 0.0001	Ci	kg	Atoms	Bq
Br-82	6.0733E-04	5.6097E-13	4.1198E+12	2.2471E+07
Br-83	1.6156E-05	1.0227E-15	7.4200E+09	5.9776E+05
Kr-83m	1.2500E-02	6.0583E-13	4.3957E+12	4.6248E+08
Kr-85	8.1467E-01	2.0751E-06	1.4702E+19	3.0143E+10
Kr-85m	1.6234E-01	1.9727E-11	1.3976E+14	6.0067E+09
Kr-87	2.6301E-05	9.2854E-16	6.4273E+09	9.7315E+05
Kr-88	5.0143E-02	3.9989E-12	2.7366E+13	1.8553E+09
Rb-87	4.4281E-24	5.0613E-20	3.5034E+05	1.6384E-13
Rb-88	1.1716E-05	9.7601E-17	6.6792E+08	4.3349E+05
I-129	6.8183E-09	3.8601E-08	1.8020E+17	2.5228E+02
I-130	1.4897E-03	7.6381E-13	3.5383E+12	5.5118E+07
I-131	1.9834E-01	1.5999E-09	7.3546E+15	7.3387E+09
I-132	1.5753E-01	1.5262E-11	6.9627E+13	5.8288E+09
I-133	1.2391E-01	1.0938E-10	4.9529E+14	4.5847E+09
I-134	6.5286E-09	2.4473E-19	1.0999E+06	2.4156E+02
I-135	2.0391E-02	5.8063E-12	2.5901E+13	7.5446E+08
Xe-129m	2.2597E-04	1.7859E-12	8.3372E+12	8.3607E+06
Xe-131m	2.9842E-01	3.5627E-09	1.6378E+16	1.1041E+10
Xe-133	5.0066E+01	2.6747E-07	1.2111E+18	1.8525E+12
Xe-133m	1.5255E+00	3.3997E-09	1.5394E+16	5.6442E+10
Xe-135	1.3783E+01	5.3973E-09	2.4076E+16	5.0998E+11
Xe-135m	6.5313E-01	7.1700E-12	3.1984E+13	2.4166E+10
Cs-135	4.7418E-14	4.1157E-14	1.8359E+11	1.7545E-03

Control Room Transport Group Inventory:

Time (h) = 0.0001	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5969E+19	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.8787E+17	0.0000E+00	0.0000E+00
Organic I (atoms)	2.8223E+14	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 0.0001	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 0.0001	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 0.0001	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 2.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.0214E-03	0.0000E+00	3.4587E-03
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 2.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.1402E-04	0.0000E+00	3.8609E-0
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 2.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	4.3741E-02	0.0000E+00	1.8080E+00
Accumulated dose (rem)	4.3742E-02	0.0000E+00	1.8080E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 2.0000	Ci	kg	Atoms	Bq
Br-82	5.8394E-04	5.3937E-13	3.9611E+12	2.1606E+07
Br-83	9.0455E-06	5.7258E-16	4.1544E+09	3.3468E+05
Kr-83m	5.8665E-03	2.8434E-13	2.0631E+12	2.1706E+08
Kr-85	8.1466E-01	2.0750E-06	1.4701E+19	3.0143E+10

Kr-85m	1.1914E-01	1.4477E-11	1.0257E+14	4.4081E+09
Kr-87	8.8422E-06	3.1216E-16	2.1608E+09	3.2716E+05
Kr-88	3.0777E-02	2.4545E-12	1.6797E+13	1.1388E+09
Rb-87	5.3930E-20	6.1643E-16	4.2669E+09	1.9954E-09
Rb-88	3.4843E-02	2.9027E-13	1.9864E+12	1.2892E+09
I-129	6.8183E-09	3.8601E-08	1.8020E+17	2.5228E+02
I-130	1.3316E-03	6.8277E-13	3.1629E+12	4.9271E+07
I-131	1.9692E-01	1.5884E-09	7.3020E+15	7.2861E+09
I-132	8.6223E-02	8.3532E-12	3.8109E+13	3.1902E+09
I-133	1.1592E-01	1.0233E-10	4.6335E+14	4.2892E+09
I-134	1.3431E-09	5.0346E-20	2.2626E+05	4.9693E+01
I-135	1.6533E-02	4.7078E-12	2.1001E+13	6.1173E+08
Xe-129m	2.2434E-04	1.7731E-12	8.2773E+12	8.3006E+06
Xe-131m	2.9698E-01	3.5456E-09	1.6299E+16	1.0988E+10
Xe-133	4.9536E+01	2.6464E-07	1.1983E+18	1.8328E+12
Xe-133m	1.4858E+00	3.3113E-09	1.4993E+16	5.4975E+10
Xe-135	1.1852E+01	4.6412E-09	2.0704E+16	4.3854E+11
Xe-135m	5.5597E-03	6.1033E-14	2.7226E+11	2.0571E+08
Cs-135	8.8064E-10	7.6435E-10	3.4096E+15	3.2584E+01

Control Room Transport Group Inventory:

Time (h) = 2.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5969E+19	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.8787E+17	0.0000E+00	0.0000E+00
Organic I (atoms)	2.8223E+14	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 2.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 2.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 2.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 8.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 8.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 8.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	9.7343E-02	0.0000E+00	4.9289E-01
Accumulated dose (rem)	1.4108E-01	0.0000E+00	2.3009E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 8.0000	Ci	kg	Atoms	Bq
Br-82	9.2289E-10	8.5245E-19	6.2604E+06	3.4147E+01
Kr-83m	6.0467E-04	2.9307E-14	2.1264E+11	2.2373E+07
Kr-85	8.1463E-01	2.0749E-06	1.4701E+19	3.0141E+10
Kr-85m	4.7085E-02	5.7215E-12	4.0536E+13	1.7422E+09
Kr-87	3.3592E-07	1.1859E-17	8.2090E+07	1.2429E+04
Kr-88	7.1163E-03	5.6752E-13	3.8837E+12	2.6330E+08
Rb-87	3.4993E-22	3.9998E-18	2.7686E+07	1.2948E-11
Rb-88	4.0916E-03	3.4086E-14	2.3326E+11	1.5139E+08
I-129	1.2123E-14	6.8636E-14	3.2041E+11	4.4857E-04
I-130	1.6912E-09	8.6715E-19	4.0170E+06	6.2576E+01
I-131	3.4268E-07	2.7641E-15	1.2707E+10	1.2679E+04
I-132	2.5135E-08	2.4350E-18	1.1109E+07	9.2999E+02
I-133	1.6877E-07	1.4898E-16	6.7457E+08	6.2443E+03
I-135	1.5670E-08	4.4619E-18	1.9904E+07	5.7977E+02
Xe-129m	2.1953E-04	1.7351E-12	8.0999E+12	8.1227E+06
Xe-131m	2.9269E-01	3.4944E-09	1.6064E+16	1.0830E+10
Xe-133	4.7973E+01	2.5629E-07	1.1605E+18	1.7750E+12
Xe-133m	1.3727E+00	3.0592E-09	1.3852E+16	5.0790E+10
Xe-135	7.5012E+00	2.9373E-09	1.3103E+16	2.7754E+11
Xe-135m	1.5232E-08	1.6722E-19	7.4592E+05	5.6359E+02
Cs-135	1.2451E-10	1.0807E-10	4.8206E+14	4.6067E+00

Control Room Transport Group Inventory:

Time (h) = 8.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5969E+19	0.0000E+00	0.0000E+00
Elemental I (atoms)	3.3405E+11	0.0000E+00	0.0000E+00
Organic I (atoms)	5.0183E+08	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 8.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	1.8787E+17
Organic I (atoms)	0.0000E+00	2.8223E+14
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 8.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 8.0000	Pathway Filter

Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 24.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 24.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 24.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	1.6709E-01	0.0000E+00	1.6726E-01
Accumulated dose (rem)	3.0817E-01	0.0000E+00	2.4682E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 24.0000	Ci	kg	Atoms	Bq
Kr-83m	1.4111E-06	6.8394E-17	4.9624E+08	5.2211E+04
Kr-85	8.1453E-01	2.0747E-06	1.4699E+19	3.0138E+10
Kr-85m	3.9608E-03	4.8129E-13	3.4099E+12	1.4655E+08
Kr-88	1.4332E-04	1.1430E-14	7.8218E+10	5.3029E+06
Rb-88	9.7010E-05	8.0816E-16	5.5305E+09	3.5894E+06
Xe-129m	2.0721E-04	1.6377E-12	7.6453E+12	7.6668E+06
Xe-131m	2.8154E-01	3.3613E-09	1.5452E+16	1.0417E+10
Xe-133	4.4031E+01	2.3523E-07	1.0651E+18	1.6292E+12
Xe-133m	1.1114E+00	2.4768E-09	1.1215E+16	4.1120E+10
Xe-135	2.2144E+00	8.6714E-10	3.8682E+15	8.1934E+10
Cs-135	3.9856E-11	3.4593E-11	1.5431E+14	1.4747E+00

Control Room Transport Group Inventory:

Time (h) = 24.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5969E+19	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.5500E-04	0.0000E+00	0.0000E+00
Organic I (atoms)	2.3286E-07	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 24.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	1.8787E+17
Organic I (atoms)	0.0000E+00	2.8223E+14
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 24.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 24.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 96.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 96.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 96.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	2.2611E-01	0.0000E+00	2.2611E-01
Accumulated dose (rem)	5.3428E-01	0.0000E+00	2.6943E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 96.0000	Ci	kg	Atoms	Bq
Kr-85	8.1410E-01	2.0736E-06	1.4691E+19	3.0122E+10
Kr-85m	5.7517E-08	6.9891E-18	4.9517E+07	2.1281E+03
Xe-129m	1.5978E-04	1.2628E-12	5.8953E+12	5.9119E+06
Xe-131m	2.3640E-01	2.8224E-09	1.2975E+16	8.7470E+09
Xe-133	2.9847E+01	1.5946E-07	7.2201E+17	1.1044E+12
Xe-133m	4.2964E-01	9.5751E-10	4.3355E+15	1.5897E+10
Xe-135	9.1383E-03	3.5784E-12	1.5963E+13	3.3812E+08
Cs-135	2.2265E-13	1.9325E-13	8.6205E+11	8.2380E-03

Control Room Transport Group Inventory:

Time (h) = 96.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5969E+19	0.0000E+00	0.0000E+00
Elemental I (atoms)	1.5479E-73	0.0000E+00	0.0000E+00
Organic I (atoms)	2.3253E-76	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 96.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	1.8787E+17
Organic I (atoms)	0.0000E+00	2.8223E+14
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 96.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 96.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Exclusion Area Boundary Doses:

Time (h) = 720.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.2559E-01	0.0000E+00	1.4411E+00

Outer Boundary of the LPZ Doses:

Time (h) = 720.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	0.0000E+00	0.0000E+00	0.0000E+00
Accumulated dose (rem)	4.7507E-02	0.0000E+00	1.6087E-01

Control Room Doses:

Time (h) = 720.0000	Whole Body	Thyroid	TEDE
Delta dose (rem)	2.7824E-01	0.0000E+00	2.7824E-01
Accumulated dose (rem)	8.1252E-01	0.0000E+00	2.9725E+00

Control Room Compartment Nuclide Inventory:

Time (h) = 720.0000	Ci	kg	Atoms	Bq
Kr-85	8.1036E-01	2.0641E-06	1.4624E+19	2.9983E+10
Xe-129m	1.6795E-05	1.3274E-13	6.1967E+11	6.2141E+05
Xe-131m	5.1993E-02	6.2073E-10	2.8535E+15	1.9237E+09
Xe-133	9.7071E-01	5.1859E-09	2.3481E+16	3.5916E+10
Xe-133m	1.1375E-04	2.5352E-13	1.1479E+12	4.2089E+06

Control Room Transport Group Inventory:

Time (h) = 720.0000	Atmosphere	Sump	Overlying Pool
Noble gases (atoms)	1.5969E+19	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	0.0000E+00	0.0000E+00
Organic I (atoms)	0.0000E+00	0.0000E+00	0.0000E+00
Aerosols (kg)	0.0000E+00	0.0000E+00	0.0000E+00

Time (h) = 720.0000	Deposition Surfaces	Recirculating Filter
Noble gases (atoms)	0.0000E+00	0.0000E+00
Elemental I (atoms)	0.0000E+00	1.8787E+17
Organic I (atoms)	0.0000E+00	2.8223E+14
Aerosols (kg)	0.0000E+00	0.0000E+00

Unfiltered Environment to CR Transport Group Inventory:

Time (h) = 720.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

Control Room Exhaust Transport Group Inventory:

Time (h) = 720.0000	Pathway Filter
Noble gases (atoms)	0.0000E+00
Elemental I (atoms)	0.0000E+00
Organic I (atoms)	0.0000E+00
Aerosols (kg)	0.0000E+00

2223

##
I-131 Summary
##

Time (hr)	Containment I-131 (Curies)	Environment I-131 (Curies)	Control Room I-131 (Curies)
0.000	4.3771E-01	1.8194E+02	1.9834E-01
0.275	0.0000E+00	1.8238E+02	1.9815E-01
0.525	0.0000E+00	1.8238E+02	1.9797E-01
0.775	0.0000E+00	1.8238E+02	1.9779E-01
1.025	0.0000E+00	1.8238E+02	1.9761E-01
1.275	0.0000E+00	1.8238E+02	1.9744E-01
1.525	0.0000E+00	1.8238E+02	1.9726E-01
1.775	0.0000E+00	1.8238E+02	1.9708E-01
2.000	0.0000E+00	1.8238E+02	1.9692E-01
2.250	0.0000E+00	1.8238E+02	1.1332E-01
2.500	0.0000E+00	1.8238E+02	6.5215E-02
2.750	0.0000E+00	1.8238E+02	3.7529E-02
3.000	0.0000E+00	1.8238E+02	2.1597E-02
3.250	0.0000E+00	1.8238E+02	1.2428E-02
3.500	0.0000E+00	1.8238E+02	7.1523E-03
3.750	0.0000E+00	1.8238E+02	4.1159E-03
4.000	0.0000E+00	1.8238E+02	2.3686E-03
4.250	0.0000E+00	1.8238E+02	1.3631E-03
4.500	0.0000E+00	1.8238E+02	7.8441E-04
4.750	0.0000E+00	1.8238E+02	4.5140E-04
5.000	0.0000E+00	1.8238E+02	2.5977E-04
5.250	0.0000E+00	1.8238E+02	1.4949E-04
5.500	0.0000E+00	1.8238E+02	8.6028E-05
5.750	0.0000E+00	1.8238E+02	4.9507E-05
6.000	0.0000E+00	1.8238E+02	2.8490E-05
6.250	0.0000E+00	1.8238E+02	1.6395E-05
6.500	0.0000E+00	1.8238E+02	9.4349E-06
6.750	0.0000E+00	1.8238E+02	5.4295E-06
7.000	0.0000E+00	1.8238E+02	3.1245E-06
7.250	0.0000E+00	1.8238E+02	1.7981E-06
7.500	0.0000E+00	1.8238E+02	1.0348E-06
7.750	0.0000E+00	1.8238E+02	5.9547E-07
8.000	0.0000E+00	1.8238E+02	3.4268E-07
8.400	0.0000E+00	1.8238E+02	1.4155E-07
8.700	0.0000E+00	1.8238E+02	7.2938E-08
9.000	0.0000E+00	1.8238E+02	3.7582E-08
9.300	0.0000E+00	1.8238E+02	1.9365E-08
9.600	0.0000E+00	1.8238E+02	9.9780E-09
9.900	0.0000E+00	1.8238E+02	5.1413E-09
10.200	0.0000E+00	1.8238E+02	2.6491E-09
24.000	0.0000E+00	1.8238E+02	1.5013E-22
96.000	0.0000E+00	1.8238E+02	1.1575E-91
720.000	0.0000E+00	1.8238E+02	0.0000E+00

##
Cumulative Dose Summary
##

Time (hr)	Exclusion Area Bounda Thyroid (rem)	TEDE (rem)	Outer Boundary of the Thyroid (rem)	TEDE (rem)	Control Room Thyroid (rem)	TEDE (rem)
0.000	0.0000E+00	1.4377E+00	0.0000E+00	1.6048E-01	0.0000E+00	4.5686E-05
0.275	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.5085E-01
0.525	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	4.7825E-01
0.775	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	7.0504E-01
1.025	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	9.3124E-01
1.275	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	1.1569E+00
1.525	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	1.3819E+00
1.775	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	1.6064E+00
2.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	1.8080E+00
2.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	1.9809E+00
2.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.0823E+00
2.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.1424E+00
3.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.1789E+00
3.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2017E+00
3.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2166E+00
3.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2270E+00
4.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2347E+00
4.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2410E+00
4.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2463E+00
4.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2510E+00
5.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2555E+00
5.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2597E+00
5.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2638E+00
5.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2677E+00
6.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2716E+00
6.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2755E+00
6.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2792E+00
6.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2830E+00
7.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2866E+00
7.250	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2903E+00
7.500	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2939E+00
7.750	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.2974E+00
8.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.3009E+00
8.400	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.3065E+00
8.700	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.3105E+00
9.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.3146E+00
9.300	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.3186E+00
9.600	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.3225E+00
9.900	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.3264E+00
10.200	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.3302E+00
24.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.4682E+00
96.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.6943E+00
720.000	0.0000E+00	1.4411E+00	0.0000E+00	1.6087E-01	0.0000E+00	2.9725E+00

```
#######################################################################
                        Worst Two-Hour Doses
   Note: All of the dose locations are shown below but the
         worst two-hour dose is only meaningful for the EAB
         dose location.  Please disregard the two-hour worst
         doses for the other dose locations
#######################################################################
```

Exclusion Area Boundary

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
0.0	1.0214E-03	0.0000E+00	3.4587E-03

Outer Boundary of the LPZ

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
0.0	1.1402E-04	0.0000E+00	3.8609E-04

Control Room

Time (hr)	Whole Body (rem)	Thyroid (rem)	TEDE (rem)
0.0	4.3741E-02	0.0000E+00	1.8080E+00

FirstEnergy

DESIGN VERIFICATION RECORD

HAW 1/4/02

NOP-CC-2001-01 Rev. 00

SECTION I: TO BE COMPLETED BY DESIGN ORIGINATOR

DOCUMENT(S)/ACTIVITY TO BE VERIFIED:

Calc 3.2.15.14, Revision 0

☒ SAFETY RELATED ☐ AUGMENTED QUALITY ☐ NONSAFETY RELATED

SUPPORTING/REFERENCE DOCUMENTS

DESIGN ORIGINATOR: *(Print and Sign Name)* Harry A. Wagage Harry A. Wagage — DATE 1/3/02

SECTION II: TO BE COMPLETED BY VERIFIER

VERIFICATION METHOD *(Check one)*

☒ DESIGN REVIEW *(Complete Design Review Checklist or Calculation Review Checklist)* ☐ ALTERNATE CALCULATION ☐ QUALIFICATION TESTING

JUSTIFICATION FOR SUPERVISOR PERFORMING VERIFICATION:

The SCIENTECH manager performed this verification since this is an area for which he stays abreast of the industry positions and is known throughout the industry as an expert in this area.

APPROVAL: *(Print and Sign Name)* David A. Studley — DATE 1/4/02

EXTENT OF VERIFICATION:

The FHA calculation was completely reviewed, including the methodology, the use of the RADTRAD code, the conformance with current regulatory expectations, etc. This calculation was found to be acceptable and appropriate for inclusion with the LAR to the NRC. In addition, during the course of the calculation development, an analysis was done using an Excel spreadsheet which yielded identical results. Editorial comments and comments on the wording were provided to the preparer and have been incorporated.

COMMENTS, ERRORS OR DEFICIENCIES IDENTIFIED? ☒ YES ☐ NO

RESOLUTION: *(For Alternate Calculation or Qualification Testing only)*

Editorial comments provided and satisfactorily incorporated. Tech. comment noted on the check list regarding EDE for I-135 found acceptable as is.

RESOLVED BY: *(Print and Sign Name)* David A. Studley — DATE 1/4/02

VERIFIER: *(Print and Sign Name)* David A. Studley — DATE 1/4/02

APPROVED BY: *(Print and Sign Name)* M. Donovan — DATE 1/4/02

FirstEnergy

CALCULATION REVIEW CHECKLIST

NOP-CC-2001-04 Rev. 00

CALCULATION NO. 3.2.15.14

REV. 0

UNIT

QUESTION	NA	Yes	No	COMMENTS	RESOLUTION
REFERENCES					
1 Does the stated objective/purpose clearly describe why the calculation is being performed?		X			
2. Are applicable codes, standards, design/licensing basis documents, etc., including edition and addenda where appropriate clearly identified?		X			
3. Do the references reflect the appropriate revision?		X			
INPUTS					
4. Are design inputs clearly identified and their source documents referenced, including revision level as appropriate?		X			
5. Are the design inputs relevant, current, consistent with design/licensing bases and directly applicable to the purpose of the calculation, including appropriate tolerances and ranges/modes of operation?		X			
6. Are all design inputs retrievable? If not, have they been added as attachments?		X			
7. Are preliminary or conceptual inputs clearly identified for later confirmation as open assumptions?	X				
ASSUMPTIONS					
8. Have the assumptions necessary to perform the analysis been adequately documented?		X			
9. Is suitable justification provided for all assumptions (except those based upon recognized engineering practice, physical constants or elementary scientific principles)?		X			
10. Are all assumptions for the calculation reasonable and consistent with design/licensing bases?		X			
11. Have all open assumptions needing later confirmation been clearly identified on the Calculation cover sheet, including when the open assumption needs to be closed?	X				
12. Has a Condition Report been issued for open assumptions if required?		X			
13. Have engineering judgments been used?			X		
14. Are engineering judgments reasonable and adequately documented?	X				
METHOD OF ANALYSIS					
15. Is the method used appropriate considering the purpose and type of calculation?		X			
16. Is the method in accordance with applicable codes, standards, and design/licensing bases?		X			
IDENTIFICATION OF COMPUTER CODES (Ref: NOP-SS-1001)					
17. Have the versions of the computer codes employed in the design analysis been certified for this application?		X			
18. Are codes properly identified along with source, inputs and outputs?		X			
19. Is the code suitable for the analysis being performed?		X			
20. Does the computer model, that has been created, adequately reflect actual (or to be modified) plant conditions (e.g., dimensional accuracy, type of model/code options used, time steps, etc.)?		X			
21. Is the computer output reasonable when compared to inputs and what was expected?		X			
COMPUTATIONS					
22. Are the equations used consistent with recognized engineering practice and design/licensing bases?		X			
23. Is justification provided for any equations not in common use?	X				
24. Is the justification reasonable?	X				
25. Have adjustment factors, uncertainties, empirical correlations, etc., used in the analysis been correctly applied?		X			
26. Is the result presented with proper units and tolerance?		X			
27. Has proper consideration been given to results that may be overly sensitive to very small changes in input?		X			

FirstEnergy

CALCULATION REVIEW CHECKLIST

NOP-CC-2001-04 Rev. 00

CALCULATION NO. 3.2.15.14

REV. 0

UNIT

QUESTION	NA	Yes	No	COMMENTS	RESOLUTION
CONCLUSIONS					
28. Is the magnitude of the result reasonable when compared to inputs?		X			
29. Is the direction of trends reasonable?		X			
30. Are stated conclusions justifiable based on the calculation results?		X			
31. Are all pages sequentially numbered and marked with a valid calculation number?		X			
32. Is all information legible and reproducible?		X			
33. Have all changes in the documentation been initialed (or signed) and dated by the author of the change and all required reviewers?		X			
34. Have all calculation results stayed within existing design/licensing basis parameters?			X	This calc deviates from current design and licensing basis.	LAR being prepared to adopt this calculation for PNPP.
35. If the response to Question 34 is NO, has Licensing been notified as appropriate? (i.e. UFSAR or Tech Spec Change Request has been initiated).		X			
36. Does the calculation meet its purpose/objective?		X			
37. Has the calculation vendor used all applicable design information/requirements provided?		X			
38. Did the calculation vendor determine if the calculation was referenced in design basis documents and/or databases?		X			
39. Did the Preparer determine if the calculation was used as a reference in the UFSAR?		X			
40. If the calculation is used as a reference in the UFSAR, is a change to the UFSAR required or an update to the UFSAR Validation Database, if applicable, required?		X		USAR affected.	Once LAR is approved, the USAR will be updated. LAR includes USAR changes
41. If the answer to Question 40 is YES, have the appropriate documents been initiated?			X		See item 40.
42. Is the calculation acceptable for use?		X			
43. What checking method was used to review the calculation? Check all that apply.					
• spot check for math		X			
• complete check for math		X			
• comparison with tests					
• check by alternate method		X			
• comparison with previous calculation		X			

Review Summary: The FHA calculation, methodology, use of the code, etc. was completely reviewed and found acceptable. In addition, during the course of the calculation development, an analysis was done using an Excel spreadsheet which yielded identical results. Editorial comments provided to clarify the calculation. The EDE for I135 in the calc does not agree with FR12 (available at http://www.epa.gov/radiation/federal/docs/fgr12.pdf). The calc used a value from MACCS2 (DIN15) which is slightly higher than the FR12 value (conservative which is OK).

☒ Technical Review

Reviewer (*Print and Sign Name*) David A. Studley — Date 1/4/02

☐ Owner's Acceptance Review (*Required for calculations prepared by a vendor*)

Reviewer: (*Print and Sign Name*) N/A — Date

Approver: (*Print and Sign Name*) N/A — Date

FirstEnergy

CALCULATION REVIEW CHECKLIST

NOP-CC-2001-04 Rev. 00

CALCULATION NO. 3.2.15.14

REV. 0

UNIT Perry

QUESTION	NA	Yes	No	COMMENTS	RESOLUTION
REFERENCES 1 Does the stated objective/purpose clearly describe why the calculation is being performed?		x			
2 Are applicable codes, standards, design/licensing basis documents, etc., including edition and addenda where appropriate clearly identified?		x			
3 Do the references reflect the appropriate revision?		x			
INPUTS 4. Are design inputs clearly identified and their source documents referenced, including revision level as appropriate?		x			
5. Are the design inputs relevant, current, consistent with design/licensing bases and directly applicable to the purpose of the calculation, including appropriate tolerances and ranges/modes of operation?			x	Not necessarily consistent with design/license bases. License amendment request required.	N/A
6. Are all design inputs retrievable? If not, have they been added as attachments?		x			
7. Are preliminary or conceptual inputs clearly identified for later confirmation as open assumptions?	x				
ASSUMPTIONS 8 Have the assumptions necessary to perform the analysis been adequately documented?		x			
9. Is suitable justification provided for all assumptions (except those based upon recognized engineering practice, physical constants or elementary scientific principles)?		x			
10. Are all assumptions for the calculation reasonable and consistent with design/licensing bases?			x	See comment to Item 5	N/A
11. Have all open assumptions needing later confirmation been clearly identified on the Calculation cover sheet, including when the open assumption needs to be closed?	X				
12. Has a Condition Report been issued for open assumptions if required?	X				
13. Have engineering judgments been used?		X			
14. Are engineering judgments reasonable and adequately documented?		X			
METHOD OF ANALYSIS 15. Is the method used appropriate considering the purpose and type of calculation?		X			
16. Is the method in accordance with applicable codes, standards, and design/licensing bases?			x	See comment to item 5	N/A
IDENTIFICATION OF COMPUTER CODES (Ref: NOP-SS-1001) 17. Have the versions of the computer codes employed in the design analysis been certified for this application?		x			
18. Are codes properly identified along with source, inputs and outputs?		x			
19. Is the code suitable for the analysis being performed?		x		Code is NRC endorsed	N/A
20. Does the computer model, that has been created, adequately reflect actual (or to be modified) plant conditions (e.g., dimensional accuracy, type of model/code options used, time steps, etc.)?		x			
21. Is the computer output reasonable when compared to inputs and what was expected?		x			
COMPUTATIONS 22. Are the equations used consistent with recognized engineering practice and design/licensing bases?			x	See comment to item 5	N/A
23. Is justification provided for any equations not in common use?	x				
24. Is the justification reasonable?	x				
25 Have adjustment factors, uncertainties, empirical correlations, etc., used in the analysis been correctly applied?		x			
26. Is the result presented with proper units and tolerance?		x			
27. Has proper consideration been given to results that may be overly sensitive to very small changes in input?		x		Refer to sensitivity analyses	N/A

FirstEnergy

CALCULATION REVIEW CHECKLIST

NOP-CC-2001-04 Rev. 00

CALCULATION NO. 3.2.15.14

REV. 0

UNIT Perry

QUESTION	NA	Yes	No	COMMENTS	RESOLUTION
CONCLUSIONS					
28. Is the magnitude of the result reasonable when compared to inputs?		x			
29 Is the direction of trends reasonable?		x			
30 Are stated conclusions justifiable based on the calculation results?		x			
31 Are all pages sequentially numbered and marked with a valid calculation number?		x			
32. Is all information legible and reproducible?		x			
33 Have all changes in the documentation been initialed (or signed) and dated by the author of the change and all required reviewers?		x		Two typographical errors corrected as part of this review on page 16.	N/A
34. Have all calculation results stayed within existing design/licensing basis parameters?			x	See comment to item 5	N/A
35. If the response to Question 34 is NO, has Licensing been notified as appropriate? (i e. UFSAR or Tech Spec Change Request has been initiated)		x		License amendment request has been prepared.	N/A
36. Does the calculation meet its purpose/objective?		x			
37. Has the calculation vendor used all applicable design information/requirements provided?		x			
38. Did the calculation vendor determine if the calculation was referenced in design basis documents and/or databases?	x			FE to prepare USAR Change and ATLAS database updates	N/A
39. Did the Preparer determine if the calculation was used as a reference in the UFSAR?		x			
40 If the calculation is used as a reference in the UFSAR, is a change to the UFSAR required or an update to the UFSAR Validation Database, if applicable, required?		x			
41. If the answer to Question 40 is YES, have the appropriate documents been initiated?		x		License amendment prepared.	N/A
42. Is the calculation acceptable for use?		x		Pending NRC approval	N/A
43. What checking method was used to review the calculation? Check all that apply.	X			Reviewed inputs to calculation.	N/A
• spot check for math	X			Did not re-run RADTRAD Code.	N/A
• complete check for math	X				
• comparison with tests	X				
• check by alternate method	X				
• comparison with previous calculation	X				

Review Summary: The USAR Change Request and ATLAS database update will consider the entire amendment request content and will be processed according to procedures upon NRC amendment approval. Owner's acceptance review looked at the calculation, sensitivities, appendix, as well as the attachments for approach, inputs, and conclusions. It is noted that Appendix A to the calculation evaluates a scenario that was identified in Condition Report 01-4224 as a result of a review of the draft amendment request. The associated amendment request was reviewed to ensure that the calculation and request were consistent.

☐ Technical Review

Reviewer *(Print and Sign Name)* N/A

Date

☒ Owner's Acceptance Review *(Required for calculations prepared by a vendor)*

Reviewer: *(Print and Sign Name)* A. Widmer

Date 01-08-02

Approver: *(Print and Sign Name)* THEODORE A. HILSTON

Date 01/08/02

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	USAR Section/ Reference
1.12 - (Revision 1 - 4/74;RRRC Cat. 4)		
Instrumentation for earthquakes	PNPP design conforms to this guide with the exception of Paragraph C.4.b, Response Spectrum Recorder Frequency Range. The Perry Nuclear Power Plant Response Spectrum Recorders have a frequency range of "2 Hz to 25.4 Hz," rather than the recommended 1 Hz to 30 Hz.	3.7.4
1.13 - (Revision 1 - 12/75;RRRC Cat. 4)		
Spent fuel storage facility design basis	PNPP design conforms to this guide, WITH THE EXCEPTION OF PARAGRAPH C.4. THE INVENTORY OF RADIOACTIVE MATERIALS AVAILABLE FOR LEAKAGE ARE BASED ON THE ASSUMPTIONS GIVEN IN REGULATORY GUIDE 1.183.	~~6.5.1,~~ 9.1, 9.4.2
1.14 - (Revision 1 - 8/75)		
Reactor coolant pump flywheel integrity	Not applicable to PNPP design.	-
1.15 - (Revision 1 - 12/72;RRRC Cat. 1)		
Testing of reinforcing bars for Seismic Category I concrete structures	PNPP design conforms to this guide.	3.8.1, 3.8.3, 3.8.4, 3.8.5

LICENSING DOCUMENT

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	USAR Section/ Reference
1.24 - (Revision 0 - 3/72;RRRC Cat. 1)		
Assumptions used for evaluating the potential radiological consequences of a pressurized water reactor gas storage tank failure	Not applicable to PNPP design.	-
1.25 - (Revision 0 - 3/72;RRRC Cat. 1)	Not applicable to PNPP. See Regulatory Guide 1.183.	
Assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility for boiling and pressurized water reactors	~~PNPP design conforms to this guide with the following exceptions: a. (Regulatory Position C.1.j) filter efficiencies of 95% are used in accordance with Regulatory Guide 1.52; b. (Regulatory Position C.3.a/c) dose conversion factors and average gamma energies are taken from NRC TACT III and/or TACT 5 computer code in lieu of Table 1 and Reference 12.~~	~~6.5.1, 9.1.2, 9.4.2, 15.7.4~~
1.26 - (Revision 3 - 2/76;RRRC Cat. 1)		
Quality group classifications and standards for water-, steam- and radioactive-waste-containing components of nuclear power plants	PNPP design complies with this guide.	3.2.1, Table 3.2-1, 6.2.4, 6.5, 6.7, 9.4, 9.5, 10.3.3, 17.2

Enclosure 5
PY-CEI/NRR-2674L
Page 2 of 53

PNPP DESIGN CHAPTER

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	USAR Section/ Reference
1.163 (Continued)	The containment isolation check valves in the Feedwater penetrations are tested per the Inservice Testing Program.	
8.1 - (Revision 0 - 2/73)		
Radiation symbol	PNPP conforms to this guide.	-
1.183 - (Revision 0 - 7/2000)		
Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	PNPP conforms to this guide for the Fuel Handling Accident with the following exceptions: • Appendix B, Section 2; Water depth above reactor flange inside containment is less than 23 feet • Table 6, and Appendix B Sections 4.1 and 5.3; The radioactivity that escapes from the pool is assumed to be released to the environment instantaneously.	9.1.2 9.4.2 15.7.4 15.7.6 Tech Specs

THIS DOCUMENT IS NOT CONTROLLED

A radiological analysis, as a result of a fuel drop ~~from 10 feet~~ above the spent fuel pool racks is bounded by the analysis for a fuel drop inside containment. Sections 15.7.4 and 15.7.6 ~~has been performed. It identifies~~ identify the causes of the accident, assumptions and starting conditions. The fission product release from the fuel and the airborne activity ~~in the fuel handling area of the intermediate building and~~ released to the environs is calculated and the corresponding radiological effects offsite are evaluated using the methods, assumptions and conditions in Regulatory Guide 1.183. ~~1.25 and 1.52.~~ ~~The discussion of the consequences of this accident are presented in Section 15.7.4.~~

9.1.2.3.3 Spent Fuel Rack Design - GE Racks

Spent fuel rack design features are as follows (refer also to Figure 9.1-2):

a. Each containment spent fuel pool contains 19 sets of racks which may contain up to 190 fuel assemblies. A maximum of 380 fuel assemblies may be stored in the two spent fuel pools.

b. The storage racks provide an individual storage compartment for each fuel assembly and are secured to the pool wall through associated hardware. The fuel assemblies are stored in a vertical position with the lower tie plate engaged on a captive slot in the lower fuel rack support casting. Additional restraints are provided to restrict lateral movement.

c. The weight of the fuel assembly is held by the lower rack support casting.

d. The spent fuel storage racks are made from aluminum. Materials used for construction are specified in accordance with the latest issue of applicable ASTM specifications. The material choice is based on a consideration of the susceptibility of various metal combinations to electrochemical reaction. When considering the

INFORMATION ONLY

radiation alarm can be obtained in the control room following the leakage of less than 10 gallons. A plate out factor of 2 is taken into consideration for this evaluation.

With a reactor coolant pressure of 1,000 psi, and a flow restrictor of 1/4 inch, the initial flow rate through the break, assuming 100 percent flashing will be in excess of 10 gpm. Thus a monitor response to the potential hazard of less than 10 minutes is expected.

d. Fuel Handling Building Exhaust:

The drop of a channeled spent fuel bundle has been identified in Section 15.7.4 as a hazard for personnel in this building.

Assuming that ~~7.3~~ 7.49 x 10^2 Ci of Kr^{85} (Table 15.7-~~18~~ 34) are released and mixed instantaneously into the whole volume of the fuel handling building, the resulting concentration is expected to be 1.7 x 10^{-2} µCi/cc. This activity level is well within the range capability of the monitor.

The response time of the monitor is inversely proportional to the activity level and is expected to be negligible at the high anticipated levels which may be reached during this calculable fuel handling accident.

The particulate and iodine filters are removable for laboratory analysis to verify and identify activity levels and to provide a backup to the continual monitoring of the areas of surveillance.

INFORMATION ONLY

d. Generation of a condition that results in a consequential loss of function of a necessary containment barrier.

15.0.3.1.3 Unacceptable Results for Limiting Faults [Design Basis (Postulated) Accidents]

The following are considered to be unacceptable safety results for limiting faults (design basis accidents):

a. Radioactive material release which results in dose consequences that exceed the guideline values of 10 CFR 100 (for the ~~design basis RAST~~ alternative source term LOCA analysis, the offsite dose limit is 25 rem TEDE), and for the Fuel Handling Accident, the offsite dose limit is 6.3 rem TEDE).

b. Failure of fuel cladding which would cause changes in core geometry such that core cooling would be inhibited.

c. Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes.

d. Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required.

e. Radiation exposure to plant operations personnel in the main control room in excess of 5 rem whole body, 30 rem inhalation and 75 rem skin [5 rem TEDE for the ~~design basis~~ LOCA and Fuel Handling ~~(RAST analysis)~~ Accident analyses.]

15.0.3.2 Sequence of Events and Systems Operation

Each transient or accident is discussed and evaluated in terms of:

a. A step-by-step sequence of events from initiation to final stabilized condition.

INFORMATION ONLY

The Low-Low Set (LLS) Relief Function, armed upon relief actuation of any S/R valve; will cause a greater magnitude blowdown, in the relief mode, for certain specified S/R valves and a subsequent cycling of a single low set valve. The effect of the LLS design on reactor coolant pressure is demonstrated, in Chapter 5, on the MSIV closure event. This is considered bounding for all other pressurization events and, therefore, is not simulated in the analysis presented in this chapter.

A sensitivity study was also performed to support higher analytical limits for relief valve setpoints. The study shows an increase of 20 psi in the relief valve setpoint causes less than 20 psi increase in reactor peak pressures. However, these reactor peak pressures are still well below the ASME code limit (1,375 psig). Also, the increase of 20 psi in the relief setpoints does not have any effect on the peak surface heat flux, since all safety/relief valves open after the occurrence of MCPR during transients. Therefore, the analytical limits for relief valve setpoints in the Technical Specifications were 20 psi higher than those listed in Table 15.0-1.

The analytical limits used for the relief valve setpoints for the current reload analysis are listed on Table 15B.15.0-1 for the pressurization transients and on Table 15B.5.2-1 for the overpressurization transients. The analytical values are the basis for the deviation of the Technical Specification value.

15.0.3.5 Radiological Consequences

No changes; this page provided for context.

In this chapter, the consequences of radioactivity release during the three types of events: incidents of moderate frequency (anticipated operational transients), infrequent incidents (abnormal operational transients) and limiting faults (design basis accidents) are considered. For all events whose consequences are limiting a detailed quantitative evaluation is presented. For non-limiting events a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

INFORMATION ONLY

For limiting faults (design basis accidents) two quantitative analyses are considered:

a. The first is based on conservative assumptions for the purposes of worst case bounding of event consequences to determine the adequacy of the plant design to meet 10 CFR 100 guidelines (for the alternative source term ~~design basis RAST~~ LOCA analysis, the licensing basis limit is 25 rem TEDE). This analysis is referred to as the "design basis analysis." , and for the Fuel Handling Accident, the licensing basis limit is 6.3 rem TEDE (offsite) and 5 rem TEDE (control room).

b. The second is based on realistic assumptions to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis." The "realistic analysis" is not performed for the LOCA ~~RAST~~ analysis, or the Fuel Handling Accident.

Results for both are shown to be within NRC guidelines.

Doses resulting from the events in Chapter 15 are determined either manually or by computer code. Doses associated with Offgas System Failure (Section 15.7.1.1) are evaluated using GASPAR II (NUREG/CR-4653).[8] Time dependent releases are evaluated with the TACT computer code.[2][6] Instantaneous or "puff" type releases are evaluated by methods based on those presented in Regulatory Guide 1.3, 1.183, and NUREG-1465. The General Electric NEDO-31400 analysis (7) also is utilized in determining doses associated with a Control Rod Drop Accident (Section 15.4.9). Dose conversion factors and breathing rates are presented in Table 15.0-4.

15.0.4 NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA) RELATIONSHIP

Appendix 15A is a comprehensive, total plant, system-level, qualitative failure modes and effects analysis, relative to all the Chapter 15 events considered, the protective sequences utilized to accommodate the events and their effects, and the systems involved in protective actions.

INFORMATION ONLY

3. "General Electric Company Model for Loss of Coolant Analysis in Accordance with 10 CFR 50, Appendix K," December 1975 (NEDO-20566).

4. ASME Boiler and Pressure Vessel Code, Section III, Class 1, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure."

5. General Electric Company "General Electric Standard Application for Reactor Fuel" including the "United States Supplement," NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision).

6. U.S. Nuclear Regulatory Commission Computer Code TACT 5, Computer Code for Calculating Radiological Consequences of Time Varying Radioactive Releases, NUREG/CR-5106, June 1988.

7. General Electric Company "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor" NEDO-31400A, Oct. 1992.

8. U.S. NRC Computer Code GASPAR II, Computer Code to Perform Environmental Dose Analyses for Release of Radioactive Effluents. NUREG/CR-4653, March 1987.

9. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA, 1988.

10. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31897P-A, Class III (Proprietary), May 1992.

11. Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 2nd Printing, 1989

12. CCC-652 Oak Ridge National Laboratory RSICC Computer Code Collection MACCS2, V.1.12 Code Package, 1997

13. Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil", 1993

INFORMATION ONLY

TABLE 15.0-3

SUMMARY OF ACCIDENTS

Section	Title	Failed Fuel Pins: GE Calculated Value	Failed Fuel Pins: NRC Worst Case Assumption
15.3.3	Seizure of One Recirculation Pump	None	
15.3.4	Recirculation Pump Shaft Break	None	
15.4.9	Control Rod Drop Accident	<770	770
15.6.2	Instrument Line Break	None	None
15.6.4	Steam System Pipe Break Outside Containment	None	None
15.6.5	LOCA Within RCPB	None	100%
15.6.6	Feedwater Line Break	None	None
15.7.1.1	Main Condenser Offgas Treatment System Failure	N/A	N/A
15.7.3	Liquid Radwaste Tank Failure	N/A	N/A
15.7.5	Spent Fuel Cask Drop Accident	None	None
15.7.6	Fuel Handling Accident Inside Containment (GE 14 FUEL W/TRIANGULAR MAST)	~~<124~~ 151	~~124~~
15.8	ATWS	SPECIAL EVENT	

INFORMATION ONLY

TABLE 15.0-4

DOSE CONVERSION FACTORS*

Isotope	Thyroid (rem/Ci)	Whole Body 0.25xMeV/dis
I-131	1.49E+6	8.72E-2
I-132	5.35E+4	5.13E-1
I-133	3.97E+5	1.55E-1
I-134	2.54E+4	5.32E-1
I-135	1.24E+5	4.21E-1
Kr-83m		5.02E-6
Kr-85		3.72E-2
Kr-85m		5.25E-4
Kr-87		1.87E-1
Kr-88		4.64E-1
Kr-89		5.25E-1
Xe-131m		2.92E-3
Xe-133m		8.00E-3
Xe-133		9.33E-3
Xe-135m		9.92E-2
Xe-135		5.72E-2
Xe-137		4.53E-2
Xe-138		2.81E-1

Breathing Rates

Time Period (hr)	Breathing Rate ($m^{(3)}$/sec) LOCA	FHA
0-8	3.47E-4**	3.5E-4**
8-24	1.75E-4	1.8E-4
24-720	2.32E-4	2.3E-4

*The following dose conversion factors (DCF's) are used in the ~~design~~ alternative ~~basis LOCA RAST~~ source term analyses;

LOCA- DCF's for inhalation: EPA Federal Guidance Report 11 - 1988 (Reference 9)

**This breathing rate was used for the duration of the Control Room radiological consequence analyses.

FHA - CEDE: EPA Federal Guidance Report 11 - 1989 (Reference 11)
DDE/EDE: MACCS2 computer code (Reference 12), which uses Federal Guidance Report 12 - 1993 (Reference 13).

INFORMATION ONLY

k. Credit is taken for dilution in the lake to the nearest drinking water intake (0.5 miles ENE of the plant) as presented in Table 5.1-10 of the Perry Nuclear Power Plant Environmental Report (Operating License Stage).

The resulting exposures from liquid releases to the groundwater are presented in Table 15.7-14.

The individual isotopic concentrations and fraction of maximum permissible concentrations (FMPC) for the radionuclides released by a postulated failure of the concentrated waste tank are given in Table 15.7-15a. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the 10 CFR 20 regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised 10 CFR 20 dated October 4, 1993.)
A summary of the total isotopic concentration and total FMPC for each component postulated to fail is given in Table 15.7-16.

As indicated by these results the concentrations are well within the 10 CFR 20 effluent concentration limits for unrestricted areas (10 CFR 20, Appendix B). Likewise, the resultant exposures are a small fraction of acceptable limits for this type of event.

15.7.4 FUEL HANDLING ACCIDENT OUTSIDE CONTAINMENT

This accident is not reanalyzed as part of the reload analyses as the FUEL HANDLING ACCIDENT INSIDE CONTAINMENT (SEE 15.7.6) ~~initial cycle~~ analysis is bounding. Radiological exposures were recalculated ~~for Cycle 9~~ incorporating GE14 fuel resulting in exposures ~~well below 10 CFR 100 guidelines.~~ WITHIN THE LICENSING BASIS LIMITS OF 6.3 REM TEDE (OFFSITE) AND 5 REM TEDE (CONTROL ROOM)

INFORMATION ONLY

15.7.4.1 Identification of Causes and Frequency Classification

15.7.4.1.1 Identification of Causes

The fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism resulting in the dropping of a raised fuel assembly onto stored fuel bundles.

15.7.4.1.2 Frequency Classification

This event has been categorized as a limiting fault.

No changes to this page; included for information.

INFORMATION ONLY

15.7.4.2 Sequence of Events and Systems Operation

15.7.4.2.1 Sequence of Events

The most severe fuel handling accident from a radiological release viewpoint is the drop of a channeled spent fuel bundle onto unchanneled spent fuel in the spent fuel racks in the fuel handling building. The sequence of events which is assumed to occur is as follows:

	Event	Approximate Elapsed Time
a.	Channeled fuel bundle is being handled by a crane over spent fuel pool. Crane motion changes from horizontal to vertical and the fuel grapple releases, dropping the bundle. The channeled bundle strikes unchanneled bundles in the rack.	0
b.	Some rods in both the dropped and struck bundles fail, releasing radioactive gases to the pool water.	0
c.	Gases pass from the water to the fuel handling building.	0
d.	The fuel handling building ventilation system high radiation alarm alerts plant personnel.	<1 Min

15.7.4.2.1.1 Identification of Operator Actions

The accident analysis does not assume any operator actions for the mitigation of this event.

No changes to this page; included for information.

INFORMATION ONLY

15.7.4.2.2 Systems Operation

~~Normally, operating plant instrumentation and controls are assumed to function although credit is taken only for the operation of the FHAES~~. Operation of ~~other~~ plant, reactor protection or ESF systems is not taken into account.

15.7.4.2.3 The Effect of Single Failures and Operator Errors

The FHAES is designed to single failure criteria and safety requirements. NO CREDIT IS TAKEN FOR THE FHAES.

~~Refer to Sections 7.3 and 9.4, and Appendix 15A for further details~~.

INFORMATION ONLY

15.7.4.3 Core and System Performance (INITIAL CYCLE)

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a realistic, yet conservative assessment of the consequences for the initial cycle.

The kinetic energy acquired by a falling fuel assembly may be dissipated in one or more impacts.

To estimate the expected number of failed fuel rods in each impact, a conservation of energy approach is used. The fuel assembly is expected to impact on the spent fuel racks at a small angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Therefore, fuel rods are expected to absorb little energy prior to failure as a result of bending. Actual bending tests with concentrated point-loads show that each fuel rod absorbs approximately 1 ft-lb prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 250 ft-lb before cladding failure (based on 1 percent uniform plastic deformation of the rods).

The energy of the dropped assembly is conservatively assumed to be absorbed by only the cladding and other pool structures. Because an unchanneled fuel assembly consists of 76 percent fuel, 19 percent cladding and 5 percent other structural material by weight, the assumption that no energy is absorbed by the fuel material results in considerable conservatism in the mass-energy calculations that follow.

INFORMATION ONLY

The energy absorption on successive impacts is estimated by considering a plastic impact. Conservation of momentum under a plastic impact shows that the fractional kinetic energy absorbed during impact is:

$$1 - \frac{M_1}{M_1 + M_2}$$

where M_1 is the impacting mass and M_2 is the struck mass.

15.7.4.3.2 Input Parameters and Initial Conditions (INITIAL CYCLE)

The assumptions used in the analysis of this accident are listed below:

a. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment.

b. The entire amount of potential energy, referenced to the top of the spent fuel racks, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the rack and requires the complete detachment of the assembly from the fuel hoisting equipment. This is only possible if the fuel assembly handle, the fuel grapple or the grapple cable breaks.

c. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

INFORMATION ONLY

15.7.4.3.3 Results (INITIAL CYCLE)

a. Energy Available

Dropping a fuel assembly onto the spent fuel racks from the maximum assumed height of 10 ft (actual height is 8 ft), results in an impact velocity of 25.4 ft/sec.

The kinetic energy acquired by the falling fuel assembly is less than 8,000 ft-lb and is dissipated in one or more impacts.

b. Energy Loss Per Impact

Based on the fuel geometry in the spent fuel rack, two fuel assemblies are struck by the impacting assembly. The fractional energy loss on the first impact is approximately 63 percent.

The second impact is expected to be less direct. The broad side of the dropped assembly impacts approximately 22 more fuel assemblies, so that after the second impact only 88 ft-lb (approximately 2 percent of the original kinetic energy), is available for a third impact. Because a single fuel rod is capable of absorbing 250 ft-lb in compression before cladding failure, it is unlikely that any fuel rod will fail on a third impact. In calculating the activity release, however, it is conservatively assumed that one rod fails on the third impact.

If the dropped fuel assembly strikes only one fuel assembly on the first impact, the energy absorption by the fuel rack support structure results in approximately the same energy dissipation on the first impact as in the case where two fuel assemblies are

INFORMATION ONLY

struck. The energy relations on the second and third impacts remain approximately the same as in the original case. Thus, the calculated energy dissipation is as follows:

First impact	63 percent
Second impact	35 percent
Third impact	2 percent (no cladding failures)

c. Fuel Rod Failures (INITIAL CYCLE)

1. First Impact Failures

The first impact dissipates 0.63 x 8,000 or 5,040 ft-lb of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent is absorbed by the struck fuel assemblies in rack. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure and because 1 ft-lb of energy is sufficient to cause cladding failure as a result of bending, all 62 rods of the dropped fuel assembly are assumed to fail. Because the 8 tie rods of each struck fuel assembly are more susceptible to bending failure than the other 54 fuel rods, it is assumed that they fail on the first impact. Thus 2 x 8 = 16 tie rods (total in 2 assemblies) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place in the spent fuel racks, they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod as a result of compression, 250 ft-lb of energy is required. To cause failure of all the remaining rods of the 2 struck assemblies, 250 x 54 x 2 or 27,000 ft-lb of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods

INFORMATION ONLY

of the struck assemblies can fail on the first impact. The number of fuel rod failures caused by compression is computed as follows:

$$\frac{0.5 \times 5{,}040 \times \frac{19}{19 + 5}}{250} = 8$$

Thus, during the first impact, fuel rod failures are as follows:

Dropped assembly	62 rods (bending)
Struck assemblies	16 tie rods (bending)
Struck assemblies	8 rods (compression)
	86 failed rods

2. Second Impact Failures

Because of the less severe nature of the second impact and the distorted shape of the dropped fuel assembly, it is assumed that in only 2 of the 22 struck assemblies are the tie rods subjected to bending failure. Thus 2 x 8 = 16 tie rods are assumed to fail. The number of fuel rod failures caused by compression on the second impact is computed as follows:

$$\frac{\frac{0.35}{2} \times 8{,}000 \times \frac{19}{19 + 5}}{250} = 5$$

No changes to this page; provided for information.

INFORMATION ONLY

Thus, during the second impact the fuel rod failures are as follows:

Struck assemblies	16 tie rods (bending)
Struck assemblies	5 rods (compression)
	21 failed rods

3. Total Failures

The total number of failed rods resulting from the accident is as follows:

First impact	86 rods
Second impact	21 rods
Third impact	1 rods
	108 total failed rods (initial cycle)

15.7.4.3.4 RESULTS (CURRENT CYCLE)

~~151 total failed rods are assumed for CE14 fuel~~.
THE TOTAL ~~FAILED~~ NUMBER OF FAILED FUEL RODS FOR THE BOUNDING FUEL HANDLING ACCIDENT IS GIVEN IN 15.7.6.3.

15.7.4.4 Barrier Performance

This failure occurs in the fuel handling building outside the normal barriers (RCPB and containment). Therefore, this section is not directly applicable. The transport of fission products to the environment is discussed in the next section.

15.7.4.5 Radiological Consequences

~~Three separate radiological analyses are provided for this accident~~:

~~a.~~ The ~~first~~ ANALYSIS is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet ~~10 CFR 100 guidelines~~ REGULATORY GUIDE 1.183 DOSE CRITERIA. This analysis is

INFORMATION ONLY

referred to as the "design basis analysis," and is based on a 24 hour radioactive decay period of the fuel.

~~b. The second analysis is based on assumptions considered to provide a realistic conservative estimate of radiological consequences. This analysis is referred to as the "realistic analysis," and is based on a 24 hour radioactive decay period of the fuel.~~

~~c. The third analysis is an assessment of offsite and Control Room dose consequences when irradiated fuel is being handled after seven days of radioactive decay has occurred. This analysis considers both conservative design basis and realistic source terms~~. When comparing a fuel handling accident inside containment with a fuel handling accident in the Fuel Handling Building, the inside containment event would be bounding due to higher kinetic energy and the greater number of fuel pins damaged. ~~Both analyses make~~ The ~~equivalent~~ assumption IS MADE that the activity which escapes from the pool is released immediately and directly to the environment. Thus, for this analysis, refer to Section 15.7.6.4 on the bounding analyses for the fuel handling accident inside containment.

~~For all analyses,~~ The fission product inventory in the fuel rods assumed to be damaged is based on operation at 3,833 MWt.

~~15.7.4.5.1 Design Basis Analysis Assuming 24 Hour Radioactive Decay of the Fuel~~

The design basis analysis is based on ASSUMPTIONS AND METHODS CONTAINED IN ~~NRC Standard Review Plan 15.7.4 and~~ NRC Regulatory Guide ~~1.25~~ 1.183. ~~Specific values of parameters used in the evaluation are presented in Table 15.7-17~~.

INFORMATION ONLY

15.7.4.5.1.1 Fission Product Release from Fuel

REFER TO SECTION 15.7.6.4.1

The following conditions are assumed applicable for this event:

a. The fuel rod gap activity is assumed to consist of 10% of the total halogen and noble gas activity in the rods at the time of the accident, except for Kr-85 which is assumed to be 30%.

b. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be available for release.

c. It is conservatively assumed that 100 percent of the noble gas plenum activity and 1.0 percent of the halogen plenum activity in the damaged fuel rods is released from the spent fuel pool to the fuel handling building atmosphere.

INFORMATION ONLY

~~Based on the above conditions the activity airborne in the fuel handling building is presented in Table 15.7-18.~~

15.7.4.5.1.2 Fission Product Transport to the Environment

REFER TO SECTION 15.7.6.4.1

~~In accordance with the criteria presented in Regulatory Guides 1.25 and 1.52 it is assumed that the airborne activity in the fuel handling building (Table 15.7-18) is released to the environment over a 2 hour period via a 95 percent iodine efficient FHAES. The total activity released to the environment is presented in Table 15.7~~-19.

15.7.4.5.1.3 Results

The calculated exposures for the BOUNDING FUEL HANDLING ACCIDENT ~~design basis analysis~~ are presented in Table 15.7-~~20~~ 35 and are ~~well~~ within the ~~guidelines of 10 CFR 100~~. DOSE CRITERIA OF REG GUIDE 1.183.

~~15.7.4.5.2 Realistic Analysis Assuming 24 Hour Radioactive Decay of~~ the Fuel

The realistic analysis is based on a realistic but still conservative assessment of this accident. Specific values of parameters used in the evaluation are presented in Table 15.7-17.

15.7.4.5.2.1 Fission Product Release from Fuel

Fission product release estimates for the fuel handling accident are based on the following assumptions:

a. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity is in the fuel rod plena and available for release. This assumption is based on fission product release data from defective fuel experiments.[3]

INFORMATION ONLY

b. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be released.

c. It is conservatively assumed that 100 percent of the noble gas plenum activity and 1.0 percent of the halogen plenum activity in the damaged fuel rods is released from the spent fuel pool to the fuel handling building atmosphere.

Based on the above conditions the activity airborne in the fuel handling building is presented in Table 15.7-21.

15.7.4.5.2.2 Fission Product Transport to the Environment

It is conservatively assumed that all activity released to the fuel handling building is released to the environment in the first two hours after the accident via a 95 percent iodine efficient FHAES. Based on these assumptions, the activity released to the environment is shown in Table 15.7-22.

15.7.4.5.2.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.7-23 and are well below the guidelines set forth in 10 CFR 100.

15.7.4.5.3 Design Basis and Realistic Analyses Assuming Seven Day Radioactive Decay

The radiological releases from a fuel handling accident inside containment (based on a seven day decay) are larger than those from a fuel handling accident outside containment (based on a seven day decay). Therefore, refer to Section 15.7.6.4.3 for details related to the bounding analyses for a fuel handling accident inside containment.

INFORMATION ONLY

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

This accident is not affected by the reload analysis.

15.7.5.1 Cask Drop from Transport Vehicle

In the unlikely event that the fuel cask falls from the transport vehicle, the maximum height which the cask will drop should be in general less than 10 ft. Since the cask is designed to withstand a 30 ft drop onto a non-yielding surface without failure, the fall from the transport vehicle will cause no failure of the cask.

15.7.5.2 Cask Drop from Crane

The Mark III containment design includes a separate fuel handling building. The spent fuel storage pools in this building are arranged so that the overhead crane which handles the cask cannot possibly move the cask above the spent fuel storage pool. This precludes the possibility of the cask falling on the stored spent fuel bundles. Also, the pools are arranged so that a rupture of the cask loading pool floor will not drain water from the spent fuel storage pool. The cask loading area design and operating procedures are specifically formulated so that a cask drop will not result in failure of the cask.

15.7.6 FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

The radiological exposures were recalculated ~~for Cycle 9~~ incorporating GE14 Fuel and power uprate analysis resulting in exposures ~~well below 10 CFR 100 Guidelines.~~ within the licensing basis limits of 6.3 rem TEDE (offsite) and 5 rem TEDE (control room).

The analysis in Section 15.7.6.4 ~~(a)~~ assumes a ~~seven day~~ 24 hour decay time prior to the accident occurring. The ~~seven day~~ 24 hour value forms the definition of "recently irradiated fuel" (i.e., the fuel has not been in a critical reactor core within the previous 24 hours.) as identified in the Technical

INFORMATION ONLY

Specifications. This analysis is reviewed each cycle to verify that the ~~seven day~~ 24 hour assumption is valid. As a result of this review, the definition of "recently irradiated fuel" may change ~~to some value other than what is analyzed in this section.~~ and would be reflected herein.

15.7.6.1 Identification of Causes and Frequency Classification

15.7.6.1.1 Identification of Causes

Various mechanisms for fuel failure during refueling have been investigated. Procedural controls, backed up by the refueling interlocks, impose restrictions on the movement of refueling equipment and control rods, to prevent an inadvertent criticality during refueling operations. In addition, the reactor protection system is able to initiate a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned

INFORMATION ONLY

criticality tests with the reactor vessel head off. It is concluded that the accident that results in the release of significant quantities of fission products during this mode of operation with the greatest analyzed radiological consequences is one resulting from the accidental dropping of a fuel bundle onto the top of the core.

The movement of non-fuel items over irradiated fuel inside containment will be administratively controlled such that the radiological consequences associated with their accidental dropping with or without primary or secondary containment will remain bounded by that of a dropped fuel bundle.

15.7.6.1.2 Frequency Classification

This event has been categorized as a limiting fault.

15.7.6.2 Sequence of Events and Systems Operation

15.7.6.2.1 Sequence of Events

The sequence of events which is assumed to occur is as follows:

	Event	Approximate Elapsed Time
a.	THE REFUELING Channeled fuel bundle being removed from reactor vessel by crane. Fuel bundle is dropped from maximum height allowed by the refueling equipment. Fuel bundle strikes core.	0
b.	Some rods in both dropped and struck bundles fail releasing radioactive gases to pool water.	0
c.	Gases pass from water immediately to building.	0
d.	Containment vessel and drywell purge ventilation system isolates due to high radiation signal. NOT CREDITED IN THE DOSE ANALYSIS	20 sec

INFORMATION ONLY

15.7.6.2.1.1 Identification of Operator Actions

The accident analysis does not assume any operator actions for the mitigation of this event.

No changes to this page; provided for information.

INFORMATION ONLY

15.7.6.3 Core and System Performance

The methods used for this evaluation are the same as those presented in Section 15.7.4.3, FOR THE INITIAL CYCLE. THE ANALYSIS CREDITS 151 FAILED FUEL RODS BASED ON GE METHODOLOGY FOR GE14 BUNDLE AND A TRIANGULAR FUEL HANDLING MAST.

15.7.6.4 Radiological Consequences

Three separate radiological ~~analyses~~ CASES are provided for this accident:

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet ~~10 CFR 100 guidelines~~ REG GUIDE 1.183 DOSE CRITERIA. This analysis is ~~referred to as the "design basis analysis," and is~~ based on a 24 hour radioactive decay period of the fuel. THIS IS CONSIDERED THE BASE CASE AND TAKES NO CREDIT FOR ENGINEERED SAFETY FEATURES, ISOLATIONS, OR FILTRATION

INSERT #1

b. ~~The second analysis is based on assumptions considered to provide a realistic conservative estimate of radiological consequences. This analysis is referred to as the "realistic analysis," and is based on a 24 hour radioactive decay period of the fuel~~.

INSERT #2

c. ~~The third analysis is an assessment of offsite and Control Room dose consequences when irradiated fuel is being handled after seven days of radioactive decay has occurred with the assumption that the release immediately and directly releases to the environment. This analysis considers both conservative design basis and realistic source terms.~~

INSERT #2A

For all ~~analyses~~ CASES, the fission product inventory in the fuel rods assumed to be damaged is based on operation at 3,833 MWt. Specific values for parameters used ~~in the first two analyses~~ are provided in Table 15.7-~~24~~ 32. ~~Specific values for the third analysis are provided in Table 15.7-32~~.

INSERT #2B

INFORMATION ONLY

Insert #1 (15.7.6.4.b)
The second case is identical to the first case but was performed to determine the effect of control room isolation and fresh air intake. The control room dose was calculated assuming that once the available activity is introduced into the control room, the fresh air intake was isolated without any additional inleakage. At two hours, outside air purge is assumed to initiate and continue for the remainder of the 30-day dose analysis.

Insert #2 (15.7.6.4.c)
The third case is also identical to the first case but was performed to determine the effect of control room isolation and emergency recirculation filtering. The control room dose was calculated assuming that once the available activity is introduced into the control room, the fresh air intake was isolated without any additional inleakage. At two hours, the control room emergency recirculation system was assumed to initiate and continue for the remainder of the 30-day dose analysis. A filtration efficiency for the charcoal of 50% was assumed in order to be consistent with Table 15.6-14.

Insert #2A
Note: The second and third cases were performed to examine the flexibility the Control Room operators have in using ventilation, to ensure that there were no dose outliers. The results show that even if the operators take 2 hours to initiate an action, (i.e., re-initiate normal intake or utilize recirculation filtration) the doses remain below the licensing basis limits.

Insert #2B
An additional scenario was reviewed in the event a fuel bundle was dropped when travelling through the refueling shield inside containment. This reduces the amount of water coverage resulting in a lower Decontamination Factor. The resultant dose is less than the cases described above when considering that the damage is limited to the number of rods in one fuel bundle.

INFORMATION ONLY

15.7.6.4.1 Design Basis Analysis Assuming 24 Hour Radioactive Decay of the Fuel

a. Fission Product Release from Fuel

INSERT #3

~~The fission product activity released from the fuel damaged as a result of a fuel handling accident is calculated using the methods outlined in Section 15.7.4.5.1.1. A total of 124 fuel rods fail as a result of this accident. For the initial cycle, a total of 151 fuel rods fail given a core loaded with GE14 fuel.~~

INFORMATION ONLY

Insert 3 (15.7.6.4.1.a)
The fission product activity released from the fuel damaged as a result of a fuel handling accident is calculated using the methods below:

1) The fuel rod gap activity is assumed to consist of 5% of the total halogen and noble gas activity in the rods at the time of the accident, except for KR-85, which is assumed to be 10% and I-131 which is assumed to be 8%.
2) Twelve percent of the alkali metals are available for release but are retained in the water.

A total of 151 fuel rods fail as a result of the accident given a core loaded with GE14 fuel.

INFORMATION ONLY

b. Fission Product Activity. ~~Airborne in the Reactor Building.~~ RELEASED

The following assumptions and initial conditions are used in calculating the fission product activity released to the ~~reactor building.~~ ENVIRONS.

1. The iodine gap inventory is composed of ~~inorganic~~ ELEMENTAL species (99.~~75~~ 85 percent) and organic species (0.~~25~~ 15 percent).

2. The minimum water depth between the top of the damaged fuel rods and the containment pool surface is 23 feet.

3. The pool decontamination factors for the ~~inorganic~~ ELEMENTAL and organic species of iodine are ~~133~~ 500 and 1, respectively, giving an overall effective decontamination factor of ~~100~~ 200 (i.e., 99.5 ~~99~~ percent of the total iodine released from the damaged rods is retained by the pool water). This difference in decontamination factors for ~~inorganic~~ ELEMENTAL and organic iodine species results in the iodine above the fuel pool being composed of ~~75~~ 57 percent ~~inorganic~~ ELEMENTAL and ~~25~~ 43 percent organic species.

4. The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1). PARTICULATE RADIONUCLIDES ARE ASSUMED TO BE RETAINED BY THE WATER.

5. The effects of plateout and fallout are neglected.

Based on these assumptions, the activity released from the pool to the ~~reactor building~~ ENVIRONMENT is listed in Table 15.7-~~25~~ 34.

6. THE ACTIVITIES WITHIN THE FAILED RODS ARE ASSUMED TO HAVE BEEN DECAYED 24 HOURS PRIOR TO THE ACCIDENT.

7. ALL ACTIVITY RELEASED FROM THE POOL TO THE CONTAINMENT IS RELEASED DIRECTLY TO THE ENVIRONMENT, (i.e., FILTERING & CONTAINMENT INTEGRITY IS NOT CREDITED)

INFORMATION ONLY

c. ~~Fission Product Release to Environs~~

The following assumptions and initial conditions are used in calculating the fission products released to the environs:

1. The containment vessel and drywell purge system are in operation at the time the accident occurs. These systems are described in Section 9.4. It is conservatively assumed that isolation of the containment vessel and drywell purge system will occur 20 seconds after the release of fission product activity from the containment pool due to a high radiation signal in this system.

2. All activity during the first 20 seconds after the accident is assumed to be released to the environs via the containment vessel and drywell purge exhaust system filter as a "puff" release.

3. No credit is taken for filtering iodine during the first 20 seconds after the accident.

4. The activity remaining in containment is released to the environs (via the annulus exhaust gas treatment system).

Based on these assumptions, the activity released to the ~~environment is presented in Table 15.7-26.~~

INFORMATION ONLY

d. Results

Based on these assumptions, the ~~integrated whole-body doses and integrated thyroid dose~~ TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE) at the exclusion boundary, ~~and~~ low population zone AND CONTROL ROOM are summarized in Table 15.7-~~27~~ 35. The doses at these distances are ~~well below the 10 CFR 100 limits~~ WITHIN THE LICENSING BASIS LIMITS OF 6.3 REM (TEDE) (OFFSITE) & 5 REM (TEDE) (CONTROL ROOM)

~~15.7.6.4.2 Realistic Analysis Assuming 24 Hour Radioactive Decay of the Fuel~~ The remainder of this section is removed. Next page is 15.7-41b.

a. Fission Product Release from Fuel

The fission product activity released from the fuel damaged as a result of a fuel handling accident is calculated using the methods outlined in Section 15.7.4.5.2.1. As a result of this accident, 124 fuel rods are assumed to fail for the initial cycle. A total of 151 fuel rods are assumed to fail for a core loaded with GE14 fuel.

b. Fission Product Activity Released to Containment

The following assumptions and initial conditions are used in calculating the fission products released to the containment:

1. The fission product activity released to the containment will be inversely proportional to the removal efficiency of the water in the upper containment pool. Because water has a negligible effect on removal of the noble gases, noble gases are assumed to be instantaneously released from the pool to the containment.

2. The iodine activity in the fuel rod plena is composed of inorganic species (99.75 percent) and organic species (0.25 percent).

INFORMATION ONLY

~~Based on these assumptions, the activity released to the containment~~ pool and ~~to~~ the environment is given in Table 15.7-33 and Table 15.7-34 respectively.

c. Results

Based on these assumptions, the integrated whole body doses and integrated thyroid doses at the exclusion boundary, low population zone, and Control Room are summarized in Table 15.7-35. The doses at these distances are well below the 10CFR100 limits, and within the General Design Criteria 19 limit for the Control Room.

15.7.7 REFERENCES FOR SECTION 15.7

1. Nguyen, D., "Realistic Accident Analysis - The RELAC Code," October 1977, (NEDO-21142).

2. Bunch, F. D., "Dose to Various Body Organs from Inhalation or Ingestion of Soluble Radionuclides," IDO-12054, AEC Research and Development Report, TID-4500, August 1966.

3. N. R. Horton, W. A. Williams, J. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," March 1969, (APED 5756).

4. General Electric Company "General Electric Standard Application for Reactor Fuel," including the United States Supplement, NEDE-24011-P-A and NEDE-P-A-US (latest approved revision).

5. Reg. Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the purpose of evaluating Compliance with 10CFR Part 50, Appendix I, Rev. 1, October 1977.

6. REGULATORY GUIDE 1.183, ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS, JULY 2000

INFORMATION ONLY

Tables 15.7-17 through 15.7-30 are deleted.

TABLE 15.7-17

FUEL HANDLING ACCIDENT OUTSIDE CONTAINMENT PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	3,833 MWt	3,833 MWt
B. Radial peaking factor	1.7	1.6
C. Fuel damage (GE14)	151 rods	151 rods
D. Release of activity by nuclide	Section 15.7.4.5.1.1	Section 15.7.4.5.2.1
E. Iodine fractions		
(1) Organic	0.0025	0.0025
(2) Elemental	0.9975	0.9975
(3) Particulate	0	0
II. Data and assumptions used to estimate activity released		
A. Fuel handling building leak rate	100%/2 hr	100%/2 hr
B. Adsorption and filtration efficiencies		
(1) Organic iodine	95%	95%
(2) Elemental iodine	95%	95%
(3) Particulate iodine	95%	95%
C. All other pertinent data and assumptions	None	None

~~REFER TO TABLE 15.7-32 FOR BOUNDING ANALYSIS PARAMETERS~~

INFORMATION ONLY

TABLE 15.7-32

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS
(ASSUMING ~~7-DAY~~ 24 HOUR RADIOACTIVE DECAY)

	Design Basis Assumptions	~~Realistic Case Assumptions~~
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	3,833 MWt	3,833 MWt
B. ~~Burn-up~~ INSERT #6	~~Section 15.7.6.4~~	Section 15.7.6.4
C. Fuel damage (GE14)	151 rods	151 rods
D. Release of activity to containment pool by nuclide, per failed rod	Section 15.7.6.4.~~3~~1	Section 15.7.6.4.3
E. Iodine fractions - organic	~~.0025~~ .0015	.0025
- ELEMENTAL ~~inorganic~~	~~.9975~~ .9985	.9975
F. RADIAL PEAKING FACTOR	2.0	
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate	Instantaneous total release of all activity leaving pool to environment	Instantaneous total release of all activity leaving pool to environment
B. Secondary containment leak rate	N/A	N/A
C. Isolation valve closure times	N/A	N/A
D. Filtration efficiencies	N/A	N/A
E. Recirculation systems parameters (flow rates vs. time, mixing factor, etc.)	N/A	N/A

INFORMATION ONLY

Insert #6 (Table 15.7-32, Item I.B)

B. Burn-up	-Peak rod exposure is less than 62,000MWD/MT -Between 54,000 and 62,000MWD/MT, the maximum linear heat generation rate does not exceed 6.3 KW/Ft peak rod average exposure.

INFORMATION ONLY

TABLE 15.7-32 (Continued)

		Design Basis Assumptions	~~Realistic Case Assumptions~~
F.	Containment spray parameters (flow rate, drop size, etc.)	N/A	~~N/A~~
G.	Containment volumes	N/A	~~N/A~~
H.	Pool Removal (iodine partition factor)	~~100~~ 200	~~100~~
I.	All other pertinent data and assumptions	Section 15.7.6	~~Section 15.7.6~~
J.	Activity released to environment	Table 15.7-~~33~~ 34	~~Table 15.7-33~~
K.	Control Room Parameters	See below	~~N/A~~
	1. Volume (ft^3)	367,070 ft^3	
INSERT #4	~~2. Design Flow (cfm)~~	~~6,600 (Instantaneous air mixing at + 10% of design flow)~~	
	~~3. Filter Efficiency~~	~~0 (No filtered recirculation assumed)~~	

III. Dispersion Data

A.	Boundary and LPZ distances (m)	863/4002	~~863/4002~~
B.	Offsite X/Q's (Corresponding to 7 year meteorological data for 0-2 hr for SB/0-8 hr for LPZ)	4.3E-4/4.8E-5	~~4.3E-4/4.8E-5~~
C.	Control Room X/Q's (0-8 hr.)	3.5E-4 ~~See below~~	~~N/A~~
	~~(1) 0-8 hrs~~	~~3.5E-4~~	
	~~(2) 8-24 hrs~~	~~2.1E-4~~	
	~~(3) 1-4 days~~	~~1.1E-4~~	
	~~(4) 4-30 days~~	~~5.75E-5~~	

IV. Dose Data

A.	Method of dose calculation	Section 15.0.3.5	~~Section 15.0.3.5~~

INFORMATION ONLY

Insert #4 (Table 15.7-32 Item II.K)

2. Design Flow (cfm)	Intake	Exhaust (AIR PURGE)	Emergency Recirculation
Case 1	6600 (Normal +10%)	5400 (Normal –10%)	0
Case 2	6600 Isolated after activity introduced into Control Room	5400 Started after 2 hrs	0
Case 3	6600 Isolated after activity introduced into Control Room	0	27,000 (Normal –10%) Started after 2 hrs Filter efficiency = 50%

INFORMATION ONLY

TABLE 15.7-32 (Continued)

		Design Basis Assumptions	~~Realistic Case Assumptions~~
B.	Dose conversion assumptions	~~Section 15.0.3.~~5	~~Section 15.0.3.5~~
	1. Dose conversion assumptions (Offsite)	Table 15.0-4	
	2. Dose Conversion Assumptions (Control Room)	Table 15.0-4	
		~~International Commission of Radiological Protection (ICRP) 30 Dose Conversion Factors (rem/Ci)~~	~~N/A~~
	~~Isotope~~		
	~~I-131~~	~~1.10E6~~	
	~~I-132~~	~~6.30E3~~	
	~~I-133~~	~~1.80E5~~	
	~~I-134~~	~~1.10E3~~	
	~~I-135~~	~~3.10E4~~	
C.	Peak activity concentrations in containment	N/A	~~N/A~~
D.	Doses	Table 15.7-35	~~Table 15.7-3~~5

INFORMATION ONLY

TABLE 15.7-33

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
ASSUMING 7 DAY RADIOLOGICAL DECAY OF THE FUEL
DESIGN BASIS AND REALISTIC ANALYSIS
ACTIVITY RELEASED TO THE CONTAINMENT POOL (CURIES)*

Isotope	Design Basis Source Terms Activity	Realistic Source Terms Activity
I-131	2.1E+4	1.7E+3
I-132	**	**
I-133	2.0E+2	5.1E+0
I-134	**	**
I-135	2.0E-3	2.9E-5
Kr-83m	**	**
Kr-85m	7.8E-8	4.3E-9
Kr-85	9.2E+2	6.2E+2
Kr-87	**	**
Kr-88	**	**
Kr-89	**	**
Xe-131m	1.9E+2	4.8E+1
Xe-133m	1.4E+3	1.4E+2
Xe-133	2.6E+4	4.3E+3
Xe-135m	**	**
Xe-135	2.8E-1	5.8E-2
Xe-137	**	**
Xe-138	**	**

* GE14 Fuel @ 3,833 MWt

** Negligible levels of activity

INFORMATION ONLY

TABLE 15.7-34

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
ASSUMING ~~7-DAY~~ 24-HOUR RADIOLOGICAL DECAY OF THE FUEL
DESIGN BASIS ~~AND REALISTIC~~ ANALYSIS
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)*

Isotope	Design Basis Source Terms Activity		Realistic Source Terms Activity
I-129 6.27E-6			
I-130 1.37E-0			
I-131	1.62E+2	2.~~1E+2~~	1.7E+1
I-132	1.45E+2	~~**~~	**
I-133	1.14E+2	2.~~0E+0~~	5.1E-2
I-134	6.00E-6	~~**~~	**
I-135	1.28E+1	2.~~0E-5~~	2.9E-7
Kr-83m	1.15E+1	~~**~~	**
Kr-85	7.49E+2	~~9.2E+2~~	6.2E+2
Kr-85m	1.49E+2	7.~~8E-8~~	4.3E-9
Kr-87	2.42E-2	~~**~~	**
Kr-88	4.61E+1	~~**~~	**
~~Kr-89~~		~~**~~	~~**~~
Xe-129m	2.08E-1		
Xe-131m	2.74E+2	1.~~9E+2~~	4.8E+1
Xe-133m	1.40E+3	~~1.4E+3~~	1.4E+2
Xe-133	4.60E+4	2.~~6E+4~~	4.3E+3
Xe-135m	6.00E+2	~~**~~	**
Xe-135	1.27E+4	2.~~8E-1~~	5.8E-2
~~Xe-137~~		~~**~~	~~**~~
~~Xe-138~~		~~**~~	~~**~~

* GE14 Fuel @ 3,833 MWt

** ~~Negligible levels of activity~~ RETAINED COMPLETELY IN THE POOL

BR 82	5.58E-1
BR 83	1.49E-2
RB 87	0 **
RB 88	0 **
~~RB~~ Cs 135	0 **

INFORMATION ONLY

TABLE 15.7-35

24-HOUR
FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
ASSUMING ~~7-DAY~~ RADIOLOGICAL DECAY OF THE FUEL
DESIGN BASIS ~~AND REALISTIC~~ ANALYSIS
RADIOLOGICAL EFFECTS

~~DESIGN BASIS SOURCE TERMS~~

	Whole Body Dose (rem)	Inhalation Dose (rem)	Beta Skin Dose (rem)
Exclusion Area (863 Meters)	1.17E-1	4.71E+1	N/A
Low Population Zone (4,002 Meters)	1.40E-2	5.25	N/A
Control Room	1.27E-2	2.76E+1	3.37E-1

REALISTIC SOURCE TERMS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area (863 Meters)	1.87E-2	3.85
Low Population Zone (4,002 Meters)	2.10E-3	0.43
Control Room	N/A	N/A

NOTE: The above radiological effects have been updated to reflect the scaled increases associated with Power Uprate to 3,758 MWt.

INSERT #5

INFORMATION ONLY

DESIGN BASIS SOURCE TERM

	TEDE Dose (rem)	Licensing Limit (rem)
Exclusion Area (863 Meters)	1.44	6.3
Low Population Zone (4002 Meters)	0.161	6.3
Control Room:		
Case 1	1.03	5
Case 2	2.81	5
Case 3	2.97	5

INFORMATION ONLY

TABLE 15A.2-4

UNACCEPTABLE CONSEQUENCES CRITERIA

PLANT EVENT CATEGORY: DESIGN BASIS ACCIDENTS

Unacceptable Consequences

4-1 Radioactive material release exceeding the guideline values of 10 CFR 100 (for the ~~design-basis RAST~~ alternative source term LOCA analysis, the licensing basis offsite dose limit is 25 rem TEDE, and for the Fuel Handling Accident, the licensing basis offsite dose limit is 6.3 rem TEDE).

4-2[1] Failure of the fuel barrier as a result of exceeding mechanical or thermal limits.

4-3 Nuclear system stresses exceeding that allowed for accidents by applicable industry codes.

4-4 Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.

4-5 Overexposure to radiation of plant main control room personnel.

NOTE:

1. Failure of the fuel barrier includes fuel cladding fragmentation (loss-of-coolant accident) and excessive fuel enthalpy (control rod drop accident).

INFORMATION ONLY

Figure 15A.6-35 presents the different protection sequences for the control rod drop accident. As shown in Figure 15A.6-35, the reactor is automatically scrammed and isolated. For all design basis cases, the neutron monitoring, reactor protection and control rod drive systems will provide a scram from high neutron flux. After the reactor has been scrammed, core cooling is accomplished by either the RCIC or the HPCS or the normal feedwater system.

b. Event 36 - Fuel Handling Accident Outside Containment

Because a fuel-handling accident can potentially occur any time when fuel assemblies are being manipulated in the fuel handling building, this accident is considered in all operating states. Considerations include mechanical fuel damage caused by drop impact and a subsequent release of fission products. The protection sequences pertinent to this accident are shown in Figure 15A.6-36. It is important to note that the systems, structures, and components described within Figure 15A.6-36 are only credited for the fuel handling accident that involves dropping of a recently irradiated bundle onto other recently irradiated bundles.

INFORMATION ONLY

h. Event 44 - Postulated Radioactive Releases Due to Liquid Containing Tank Failures

The postulated events that could cause release of the radioactive inventory of a waste tank include a tank failure and/or an operator error. The possibility of a tank failure and consequent release rates receives primary consideration in system and component design. The tanks and piping to the first isolation valve are Safety Class 3, Quality Group C components and are designed to Seismic Category I. The concentrator waste tanks are designed to operate at atmospheric pressure and 200°F maximum temperature so the possibility of failure is considered small. A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instruction. A positive action interlock system is provided to prevent inadvertent opening of a drain valve. Should a release of liquid radioactive wastes occur, floor drain sump pumps in the floor of the radwaste building will receive a high water level alarm, activate automatically and remove the spilled liquid to a contained storage tank.

The protective sequences for this event are provided in Figure 15A.6-42. No changes to this page; provided for information.

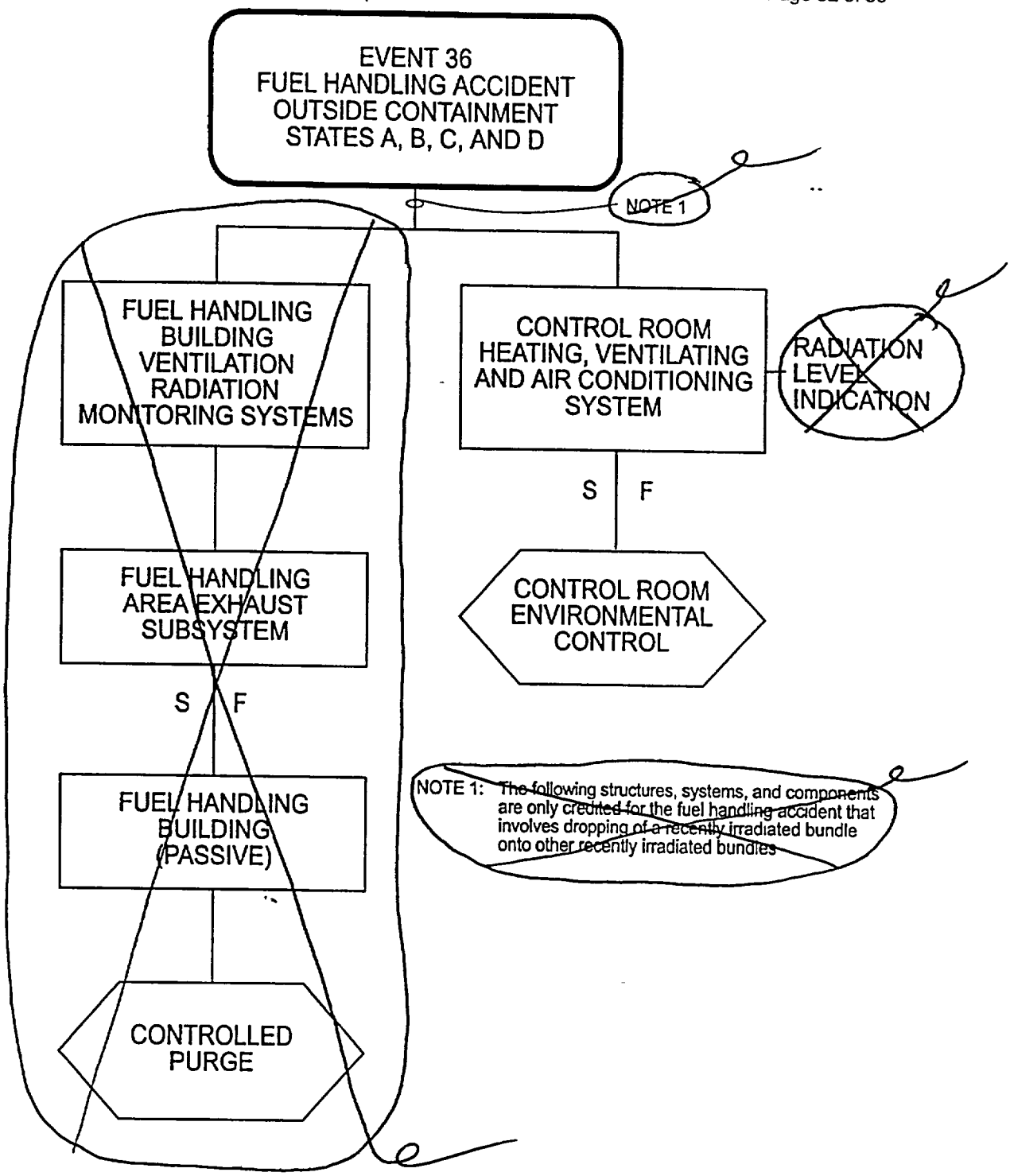
i. Event 45 - Fuel Handling Accident Inside Containment

A fuel-handling accident inside containment can only occur when fuel assemblies are being manipulated over the reactor core. Therefore, this accident is only considered in operating State A. Considerations include mechanical fuel damage caused by drop impact and a subsequent release of fission products. The protection sequences pertinent to this accident are shown in Figure 15A.6-43.

INFORMATION ONLY

~~It is important to note that the systems, structures, and components described within Figure 15A.6-43 are only credited for the fuel handling accident that involves dropping of a recently irradiated bundle onto other recently irradiated bundles.~~

INFORMATION ONLY

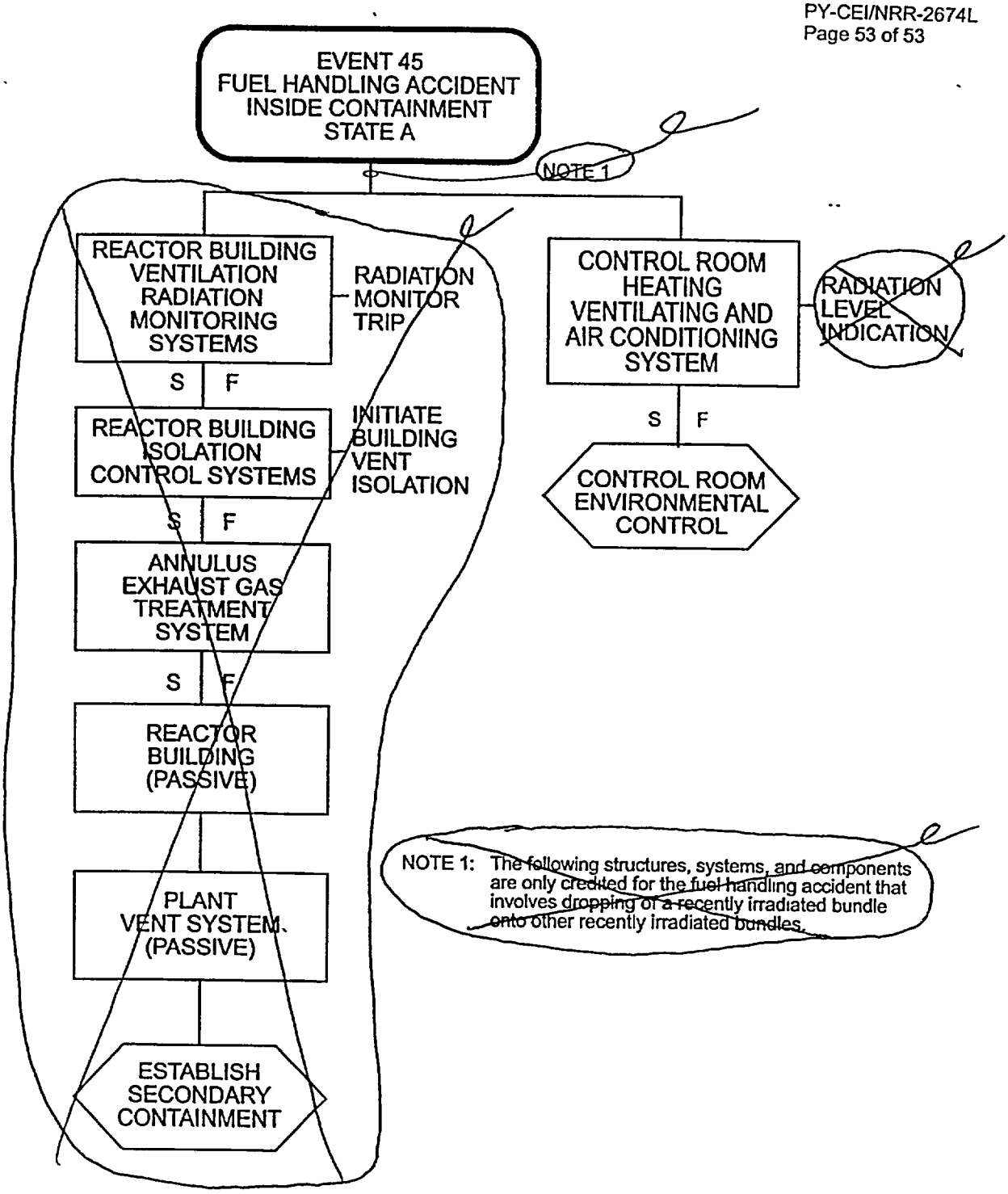


(Rev. 10 10/99)

Protective Sequences for Fuel Handling Accident Outside Containment

Figure 15A.6-36

INFORMATION ONLY



(Rev. 10 10/99)

Protective Sequences for Fuel Handling Accident Inside Containment

Figure 15A.6-43

INFORMATION ONLY