FENOC

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March 19, 2001
L-01-038

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
License Amendment Request Nos. 219 and 73

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above licenses in the form of changes to the technical specifications. The proposed amendment revises the technical specifications associated with the requirements for handling irradiated fuel assemblies in the reactor containment and in the fuel building. The proposed amendment also revises the technical specifications associated with ensuring that safety analysis assumptions are met for a fuel handling accident (FHA) in the reactor containment building and in the fuel handling building. The FHA radiological analyses that were performed to support this amendment request are based on the guidance provided in NUREG 0800 titled "Standard Review Plan" Chapter 15.0.1 and Regulatory Guide 1.183 titled "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The proposed amendment includes administrative, editorial and format changes. Revisions to the applicable Technical Specification Bases Sections are also included in this proposed amendment.

Proposed technical specification changes for Unit No. 1 are presented in Attachment A-1. Proposed technical specification changes for Unit No. 2 are presented in Attachment A-2. The safety analysis (including the no significant hazards evaluation) is presented in Attachment B. An analysis of the radiological consequences of a fuel handling design basis accident for each Unit is presented in Attachments C-1 and C-2.

The Holtec International affidavit pursuant to 10 CFR 2.790 for withholding proprietary information applicable to the Holtec International Report listed below is presented in Attachment D-1. The Proprietary version of the Holtec International Report No: HI-992343 titled "Evaluation of Spent Fuel Assembly Drop Accidents in the Beaver Valley Power Station Reactor Core" is presented in Attachment D-2. The Non-Proprietary version of the Holtec International Report No: HI-992343 titled "Evaluation of Spent Fuel Assembly Drop Accidents in the Beaver Valley Power Station Reactor Core" is presented in Attachment D-3.

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As the proprietary report contains information proprietary to Holtec International, it is supported by an affidavit signed by Holtec International, the owner of the information. The affidavit set forth the basis on which the information may be withheld from public disclosure by the Commission. Accordingly, it is respectfully requested that the information which is proprietary to Holtec International be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of this item listed above or the supporting Holtec International Affidavit should reference Holtec International Report No: HI-992343 and should be addressed to Scott H. Pellet, Holtec International, Holtec Center, 555 Lincoln Drive West, Marlton, New Jersey 08053.

Draft Containment/Fuel Building Closure Controls are present in Attachment E. Draft changes to the Licensing Requirements Manual for each Unit are presented in Attachments F-1 and F-2.

Proposed draft Updated Final Safety Analysis Report (UFSAR) changes that reflect the changes to the UFSAR description of a fuel handling accident and its radiological consequences for each Unit are presented in Attachments G-1 and G-2.

These changes have been reviewed by the Beaver Valley review committees. These changes were determined to be safe and do not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the evaluation presented in Attachment B.

An implementation period of up to 60 days is requested following the effective date of this amendment.

This change is requested to be approved prior to the start of the Beaver Valley Power Station Unit No. 1 fourteenth refueling outage (1R14).

If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager, Regulatory Affairs at 724-682-5203.

Sincerely,

Lew W. Myers

Lew W. Myers

c: Mr. L. J. Burkhart, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

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I, Lew W. Myers, being duly sworn, state that I am Senior Vice President of FirstEnergy Nuclear Operating Company (FENOC), that I am authorized to sign and file this submittal with the Nuclear Regulatory Commission on behalf of FENOC, and that the statements made and the matters set forth herein pertaining to FENOC are true and correct to the best of my knowledge and belief.

FirstEnergy Nuclear Operating Company

Lew W Myers

Lew W. Myers
Senior Vice President - FENOC

COMMONWEALTH OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 19 th day of March, 2001.

Sheila M. Fattore

My Commission Expires:

Notarial Seal
Sheila M. Fattore, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Sept. 30, 2002
Member, Pennsylv... Association of Notaries

ATTACHMENT E

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 219 and 73

DRAFT CONTAINMENT/FUEL BUILDING CLOSURE CONTROLS

BVPS ADMINISTRATIVE CLOSURE CONTROLS DURING FUEL MOVEMENT

Containment Building Closure:

A. The following requirements shall be maintained to ensure defense-in-depth:

NOTE: 1. Closure Controls are in effect whenever the Containment is open during operations within containment involving movement of non-recently irradiated fuel assemblies or movement of fuel assemblies over non-recently irradiated fuel assemblies.

2. The definition of an open containment penetration is a penetration that provides direct access from the containment atmosphere to the outside environment.

1. The equipment necessary to implement containment closure shall be appropriately staged prior to maintaining any containment penetration open including airlock doors and the containment equipment hatch.

2. Hoses and cables running through any open penetration, airlock, or equipment hatch should be tagged to facilitate rapid removal in the event that containment closure is required. The tags shall contain the following information:

   a. Directions for isolating the line prior to disconnecting.

   b. Directions for disconnecting the line.

   c. Where to perform the de-energization or isolation function.

   d. The location of any tools required for disconnection.

3. The containment personnel airlock (PAL) may be open during movement of irradiated fuel within the containment or during movement of fuel assemblies over irradiated fuel assemblies, except when the irradiated fuel has been recently irradiated (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), provided the following conditions exist:

   a. One door in each airlock is capable of being closed.

   b. Hoses and cables running through the airlock shall employ a means to allow safe, quick disconnection or severance. These hoses and cables shall be tagged per A.2 above. The tag shall be located on the hose or cable at the airlock, preferably outside the containment so it will be available for reference during containment closure.

c. The airlock door is not blocked in such a way that it cannot be expeditiously closed. Protective covers used to protect the seals/airlock doors or devices to keep the door open/supported (i.e., jacks), do not violate this provision provided these items can be removed with simple hand tools.

d. Personnel are available and designated by the Nuclear Shift Supervisor/Outage Director with the responsibility for expeditious closure of at least one door on the PAL following containment evacuation due to a fuel handling accident inside containment.

4. The containment equipment hatch may be open during movement of irradiated fuel within the containment or during movement of fuel assemblies over irradiated fuel assemblies, except when the irradiated fuel has been recently irradiated (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), provided the following conditions exist:

   a. The containment equipment hatch is capable of being closed

   b. Hoses and cables running through the equipment hatch shall employ a means to allow safe, quick disconnection or severance. These hoses and cables shall be tagged per A.2 above. The tag shall be located on the hose or cable at the equipment hatch.

   c. The equipment hatch is not blocked in such a way that it cannot be expeditiously closed. Protective covers used to protect the seals/equipment hatch or devices to keep the hatch open/flange supported (i.e., jacks), do not violate this provision provided these items can be removed with simple hand tools.

   d. 1) Necessary tools to install the equipment hatch flange and tighten at least four equipment hatch closure bolts are staged at the equipment hatch or

2) Other methods to close the equipment hatch (i.e., restrict air flow out of the containment), such as an air curtain, are fabricated and staged at the work area along with the necessary installation tools.

e. Sufficient number of personnel are available and designated by the Nuclear Shift Supervisor/Outage Director with the responsibility for expeditious closure of the containment equipment hatch following containment evacuation due to a fuel handling accident inside containment.

5. Other containment penetrations may be open during movement of irradiated fuel within the containment or during movement of fuel assemblies over irradiated fuel assemblies, except when the irradiated fuel has been recently irradiated (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours), provided the following conditions exist:

   a. One penetration isolation valve in each open containment penetration is capable of being closed, or

   b. Other methods to close the open penetrations (i.e., restrict air flow out of the containment), such as a closure cover, shall be fabricated and staged at the work area along with the necessary installation tools.

   c. Personnel are available and designated by the Nuclear Shift Supervisor/Outage Director with the responsibility for expeditious closure of open penetration(s) following a fuel handling accident inside containment.

6. Consideration should be given to maintaining containment closed during higher risk evolutions.

7. If containment closure would be hampered by an outage activity, the Outage Organization shall determine if compensatory actions are required.

8. Major disassembly of containment boundary valves, except those valves 3/4 of an inch or less in diameter, should only be performed on one valve at a time with administrative controls established on the opposite boundary valve. If conditions require working both containment isolation valves in parallel, closure devices shall be fabricated and staged at the work area.

9. The following ventilation system, with associated radiation release monitoring, will be available for the release path, whenever movement of irradiated fuel or movement of fuel assemblies over irradiated fuel assemblies is in progress in the containment building:

   a. Containment Purge and Exhaust System with either a filtered or an unfiltered exhaust path in service and the capability of placing the filtered release path in service if required, or

   b. Contiguous containment areas containing open containment penetrations, except for the containment equipment hatch, are being serviced by SLCRS.

   If for any reason the above ventilation requirements can not be met, fuel movement within the containment building shall be discontinued until the flow path(s) can be reestablished.

   The containment purge exhaust penetration will be in service in order to ensure that containment air will be drawn past at least one radiation monitor via the PAB ventilation release stack or via the SLCRS elevated release stack on top of the containment building. The automatic isolation of the purge exhaust penetration will be defeated in order to ensure that this flow path remains available should a FHA occur within containment. The capability of directing the containment purge exhaust flow through the SLCRS filter banks must be maintained.

   If for any reason operation of the purge exhaust flow path must be discontinued during fuel movement within the containment with the equipment hatch open, the equipment hatch opening will be monitored for radioactive releases via health physics air monitoring station. In addition, if other containment penetrations are open with the purge exhaust penetration not in service, the contiguous area where the open penetration is located will be verified to be exhausting via a monitored flow path or the penetration will be isolated.

10. Personnel responsible for Containment Building Closure shall be trained and knowledgeable in using the procedure for executing containment closure. Walkdowns should be considered to demonstrate the closure capability including compensatory actions in the event of loss of electrical power.

Fuel Building Closure:

B. The following requirements shall be maintained to ensure defense-in-depth:

NOTE: Closure Controls are in effect during operations within the fuel building involving movement of non-recently irradiated fuel assemblies or movement of fuel assemblies over non-recently irradiated fuel assemblies.

1. The fuel building doors shall be maintained closed except for normal entry and exit unless a designated person is available to close the open door(s) should a FHA occur within the fuel building.

2. The following ventilation system, with associated radiation release monitoring, will be available for the release flow path:

   a. The fuel building portion of SLCRS will be operated to ensure that the fuel building air is drawn past at least one radiation monitor via the elevated release stack on top of the containment. The capability of directing the fuel building exhaust flow through the SLCRS filter banks must be maintained. If for any reason operation of the fuel building SLCRS flow path must be discontinued, fuel movement within the fuel building shall be discontinued until the flow path can be reestablished.

3. If fuel building closure would be hampered by an outage activity, the Outage Organization shall determine if compensatory actions are required.

ATTACHMENT F-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 219

Applicable Draft LRM Changes

Provided for Information ONLY

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENET. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
87	H2 Discharge to CNMT		N/A	1HY-111 1HY-197	N/A N/A
88	H2 Discharge to CNMT		N/A	1HY-110 1HY-196	N/A N/A
89	Main Condenser Ejector Vent	1AS-278	N/A	(B)TV-1SV-100A	< 60
90	CNMT Purge Exhaust	(11)VS-D-5-3B	(5)8	(11)VS-D-5-3A	(5)8
91	CNMT Purge Supply	(11)VS-D-5-5B	(5)11	(11)VS-D-5-5A (11)VS-D-5-6	(5)8 N/A
92	CNMT Vacuum Pump 1B & H2 Recomb. Suction			(A)TV-1CV-150C (A)TV-1CV-150D 1HY-102 1HY-104	< 60 < 60 N/A N/A
93	CNMT Vacuum Pump 1A & H2 Recomb. Suction			(A)TV-1CV-150A (A)TV-1CV-150B 1HY-101 1HY-103	< 60 < 60 N/A N/A
94	CNMT Vacuum Ejector Suction	(11)HCV-1CV-151	N/A	(11)HCV-1CV-151-1	N/A
95	RVLIS (3 lines)			(2)(13)	N/A
95-64	H2 Analyzer - CNMT Dome	SOV-1HY-102B1	N/A	SOV-1HY-102B2	N/A
95-69	H2 Analyzer - PRZR Cubicle	SOV-1HY-103B1	N/A	SOV-1HY-103B2	N/A

TABLE 5.1-1 (Cont.)
CONTAINMENT PENETRATIONS

PENET. No.	IDENTIFICATION DESCRIPTION	INSIDE VALVE	MAXIMUM STROKE TIME (SEC)	OUTSIDE VALVE	MAXIMUM STROKE TIME (SEC)
Emergency Containment Airlock PH-P-2					
	Equalization Valve	(1)(7)1VS-184	N/A		
	Equalization Valve			(1)(7)1VS-183	N/A

NOTES:

(A) Containment Isolation Phase A
(B) Containment Isolation Phase B

(1) May be opened on an intermittent basis under administrative control.

(2) Not subject to Type C leakage tests.

(3) Tested individually by Type C Test. Leakage rates added to Air Lock Type B Test.

(4) Maximum opening time.

(5) Applicability: ~~During CORE ALTERATIONS or movement of irradiated fuel within containment~~ whenever Technical Specification 3.9.9 is applicable. In Modes 1, 2, 3 and 4 stroke time is "N/A." The provisions of Technical Specification 3.0.4 are not applicable.

(6) Not subject to the requirements of Technical Specification 3/4.6.1 and 3/4.6.3. Listed in TABLE 5.1-1 for information only.

(7) Tested under Type "B" testing.

(8) Subject to testing as per Technical Specification Amendment 65.

(Proposed Wording)

7.1 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LICENSING REQUIREMENT

LR 7.1 Loads in excess of ~~3000~~ 2450 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above LR not satisfied, place the crane load in a safe condition.

LICENSING REQUIREMENT SURVEILLANCES

LRS 7.1.1 Crane interlocks and physical stops which prevent crane travel with loads in excess of ~~3000~~ 2450 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

(Proposed Wording)

B.7.1 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the normal weight of a fuel assembly over other fuel assemblies ensures that no more than the contents of those fuel assembly rods assumed in the fuel handling accident described in Chapter 14 of the BVPS Unit 1 UFSAR will be ruptured. This assumption is consistent with the activity release assumed in the accident analyses.

and control rod assembly and associated handling tool

(Proposed Wording)

ATTACHMENT F-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 73

Applicable Draft LRM Changes

7.1 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LICENSING REQUIREMENT

2450

LR 7.1 Loads in excess of ~~3000~~ pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above LR not satisfied, place the crane load in a safe condition.

LICENSING REQUIREMENT SURVEILLANCES

LRS 7.1.1 Crane interlocks and physical stops which prevent crane travel with loads in excess of ~~3000~~ pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

2450

(Proposed Wording)

B.7.1 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the normal weight of a fuel assembly over other fuel assemblies ensures that no more than the contents of ~~one fuel assembly plus an additional 50 rods in the struck fuel assembly will be ruptured in the event of a fuel handling accident.~~ This assumption is consistent with the activity release assumed in the accident analyses.

those fuel assembly rods assumed in the fuel handling accident described in Chapter 15 of the BVPS Unit 2 UFSAR will be ruptured.

and control rod assembly and associated handling tool

(Proposed Wording)

ATTACHMENT G-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 219

Applicable Draft UFSAR Changes

The Nuclear Steam Supply System (NSSS) was designed for a warranted power output of 2,660 MWt, which is the license application rating, with an equivalent unit net electrical output of 835 MWe. The NSSS output was based upon an expected ultimate output of approximately 2,774 MWt. This rating of 2,774 MWt resulted from a core power of 2,766 MWt and 8 MWt from the reactor coolant pumps. All safety systems, including containment and engineered safety features, were designed and evaluated for operation at the higher power level.

The initial fuel load commenced in February, 1976 and commercial operation was achieved in September, 1976.

The Original FSAR consists of 15 sections and four appendices. The Updated FSAR also consists of 15 sections and four appendices, corresponding to the Original FSAR arrangement and format. The contents are briefly described below.

Section 1 of this report summarizes the principal design features and safety criteria of the unit, emphasizing the similarities and differences with respect to other pressurized water nuclear power stations employing essentially the same technology and basic engineering features as BVPS-1. Appendix 1A of Section 1 provides a discussion of BVPS-1's degree of conformance to the General Design Criteria (GDC) Published as Appendix A to 10CFR50 in July 1971. Section 2 contains a description and evaluation of BVPS-1 site and environs, prepared at the time of Plant Licensing, and supports the suitability of the site for a reactor of the size and type described. Sections 3 and 4 describe the reactor and the reactor coolant system. Section 5 describes the containment structure and related systems. Sections 6 through 11 describe the other auxiliary systems. Sections 5, 6, 7, 8, and 9 include descriptions of the various systems directly related to safety. Section 12 reviews the current organization and technical competence of BVPS-1. Descriptions of the BVPS-1 organization and personnel training are also in Section 12. Section 13 describes the initial tests and operation. Section 14 relates to safety evaluation and summarizes the analyses which demonstrate the adequacy of the reactor protection system, the containment, and engineered safety features. Section 14 demonstrates that the consequences of various postulated accidents are within the guidelines suggested in the Federal Regulation 10CFR100. Section 15, Technical Specifications and Bases has been deleted from the Updated FSAR since this section has been superceded by the BVPS-1, Technical Specifications, Appendix A to Operating License No. DPR-66. The Technical Specifications, gives safety limits, limiting safety system setting, limiting conditions for operation, surveillance requirements, design features and administrative controls for the station. The BVPS-1 Valve Operating Diagram Drawings (VOND's) provided in Section 15 of the Original FSAR have been retained to locate previously submitted documents in one location.

or 10CFR50.67, as applicable

The systems provided are summarized below:

1. The steel-lined concrete containment structure provides a highly reliable barrier against the escape of radioactivity when the containment is below or above atmospheric pressures. The structure and all penetrations, including access openings and ventilation ducts, are of proven design.

2. The emergency core cooling system cools the core by injecting borated water into the reactor coolant loops from the accumulators and the safety injection pumps in major loss-of-coolant accidents.

3. The quench spray and recirculation spray subsystems of the containment depressurization system provide sprays of borated water to the containment atmosphere. Following the DBA, the containment pressure is rapidly reduced to subatmospheric pressure by the containment depressurization system, thereby positively terminating leakage to the atmosphere. Subsequent long-term maintenance of subatmospheric conditions is accomplished by the recirculation spray subsystems and the containment vacuum system.

4. The supplementary leak collection and release system ensures that radioactive leakage from the containment penetrations following a DBA, ~~or radioactive release due to a fuel handling accident,~~ or radioactive material released in the waste gas storage area is filtered and discharged to the atmosphere at an elevated point, rather than at ground level.

5. The post DBA hydrogen control system recirculates the containment atmosphere through a recombiner following a DBA. Recombination ensures that there is no explosion or fire hazard due to hydrogen in the containment atmosphere following a DBA.

provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified on:

1. The basis of 10CFR20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur

2. The basis of 10CFR100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Answer

Provision is included in the station design for storage and processing of radioactive waste and the release of such wastes under controls adequate to prevent exceeding the limits of 10CFR20. BVPS-1 also includes provision to prevent radioactivity releases during accidents from exceeding the limits of 10CFR100. Descriptions of the radioactive waste disposal systems are included in Section 11. The effects of potential accidents, including a loss-of-coolant accident, are analyzed in Section 14.

or 10CFR50.67, as applicable

1.3.3 Safety Guides

The AEC Safety Guides applicable to BVPS-1 design and construction are provided below. Following each Safety Guide is a summary discussion of BVPS-1's method of satisfying each regulatory position.

1.3.3.1 Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)

BVPS-1 complies with the intent but not the letter of Safety Guide 1.

The regulatory position expressed in Safety Guide 1 is similar to part of the design bases as presented in Sections 6.3 and 6.4.

The operation of the emergency core cooling system is not dependent on containment pressure until after the refueling water storage tank is empty. By the time recirculation safety injection is required, the net positive suction head (NPSH) available to the recirculation pumps is sufficient to ensure satisfactory performance under all conditions. The recirculation sprays are started by containment pressure and are only required if an increase in containment pressure occurs. Consequently, Probabilities have been established by the use of general failure data based on continuous operation. Specific probability analyses will be provided on a plant basis at the request of the commission.

3. "The equipment can routinely be tested when the reactor is shut down."

In all the cases discussed above, it is only the device function that is not tested. The logic associated with the devices has the capability for testing at power.

Refer to Sections 7.2 and 7.3 for further discussion.

1.3.3.23 Onsite Meteorological Programs (Safety Guide 23)

The BVPS-1 onsite meteorological program complies with Regulatory Guide 1.23 as described in Section 2.2.3.

1.3.3.24 Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24)

The assumptions used for evaluating the potential radiological consequences of radioactive gas storage tank ruptures are provided in Section 14.2.3.

1.3.3.25 Deleted ~~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 (Safety Guide 25)~~

~~The assumptions used for evaluating the potential radiological consequences of a fuel handling accident are provided in Section 14.2.1.~~

1.3.3.26 Quality Group Classifications and Standards (Safety Guide 26)

Components for BVPS-1 were classified by quality assurance categories, as discussed in Appendix A.1.

Compliance with GDC 1 is discussed in Appendix 1A.1. Design and fabrication criteria for the Engineered Safety Feature (ESF) equipment is covered in Section 6.2. The codes and standards applicable to other systems and components are discussed within the respective sections.

1.3.3.27 Ultimate Heat Sink (Safety Guide 27)

The ultimate heat sink of BVPS-1 is the Ohio River. The river is the water source of the cooling water system that removes residual heat after reactor shutdown and following an accident.

1A.60 CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT (CRITERION 60)

Criterion

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Design Conformance

In all cases, the design for radioactivity control is justified (1) on the basis of 10CFR20 and 10CFR50 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur, and (2) on the basis of 10CFR100 dosage level guidelines for potential accidents of exceedingly low probability of occurrence.
or 10CFR50.67, as applicable,

Control of waste gas effluents is accomplished by charcoal delay beds and holdup of waste gases in decay tanks until the activity of tank contents and existing environmental conditions permit discharges within 10CFR20 and 10CFR50 requirements. In addition, waste gas effluents are monitored prior to discharge for radioactivity and rate of flow. An accidental burst of the gas surge tank does not result in an activity release greater than 10CFR100 limits, based on one percent failed fuel.

Control of liquid waste effluents is maintained by batch processing of all station radioactive liquids, sampling before discharge, controlling the rate of release, and by preventing inadvertent tank discharge. Liquid effluents are monitored for radioactivity and rate of flow. Liquid waste disposal system tankage and evaporator capacity is sufficient to handle any expected transient in the development of liquid waste volume.

Station solid wastes are prepared batchwise for offsite disposal by approved contractors. Solid wastes are prepared for shipment by placement in shielded and reinforced containers which meet Federal Regulation requirements.

References

1. Section 5, Containment System

2. Section 9, Auxiliary and Emergency Systems

defined by fuel enrichment vs. burnup limitations. The racks, free standing on the floor of the spent fuel pool, are sized to hold 1622 spent fuel assemblies (additionally there are 2 failed fuel assembly cannisters). The spent fuel assemblies are placed in vertical cells within the rack, continuously grouped in parallel in both directions. Cell pitch is approximately 10.8" for Region 1 and approximately 9" for Region 2. In addition, Boral panels are installed in the walls of the individual cells to maintain subcriticality. The racks are so arranged that the spacing between fuel elements cannot be less than that prescribed. Borated water (approximately 2000 ppm) is used in the spent fuel pool. Even if unborated water were introduced, the spacing and Boral maintain subcriticality with $K_{eff} \leq 0.95$ for stored fuel.

The new fuel assemblies are stored dry in a steel and concrete structure within the fuel building. The assemblies are stored vertically in racks in parallel rows, having a fuel assembly center-to-center distance of about 21 inches. There is storage space for one-third (53 assemblies) of a core plus 17 spare assembly spaces. The steel rack construction prevents possible criticality by requiring that the spacing between fuel elements will not be less than that prescribed. In the event of accidental flooding of the fresh fuel racks, the center to center spacing of the fuel assemblies results in $K_{eff} \leq 0.95$ under full water density conditions and $K_{eff} \leq 0.98$ under low water density (optimum moderation) and aqueous foam conditions. Criticality prevention is discussed in detail in Section 9.12 and Section 3.3.2.7.

During handling, as a result of the hypothetical worst case accident the safeguards are designed such that the consequences of this accident meet ~~10CFR100~~ 10CFR50.67 guidelines. For a complete description of this worst case accident, refer to Section 14.2.

1A.63 MONITORING FUEL AND WASTE STORAGE (CRITERION 63)

Criterion

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.

Design Conformance

Gamma radiation levels in the containment and fuel storage areas are continuously monitored. These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. The fuel pool water temperature is continuously monitored. The temperature is displayed in the main control room where an audible alarm will sound should the water temperature increase

TABLE 5.3-1 (CONT'D)

CONTAINMENT ISOLATION ARRANGEMENTS

Service	Penet. No.	Penet. Class	Nominal Line Size	Isolation Valves Provided (16) Inside	Isolation Valves Provided (16) Outside	Isolation Valve Position Normal	Shutdown (13)	DBA (14)	Failure (15)	Fluid Contained	Auto Actuation Signal (1)	1971 GDC Or Exception Met (2)	Valve Closure Time (Sec) (5) Inside	Valve Closure Time (Sec) (5) Outside	Valve Type (3) Inside	Valve Type (3) Outside	Power Source (4) Inside	Power Source (4) Outside	FSAR Fig. No.
Personnel Air Lock (PH-P-1)	None	D	1 1/2	2 Manual in Series	2 Manual in Series	Admin. Closed	Closed	Closed	AS-IS	Air	None	56	Manual	Manual	Ball	Ball	Manual	Manual	5.2-23
Emergency Air Lock (PH-P-2)	None	D	2	Manual	Manual	Admin. Closed	Closed	Closed	AS-IS	Air	None	56	Manual	Manual	Ball	Ball	Manual	Manual	5.2-23
Reactor Vessel Level Instrumentation System (RVLIS)	95,109	D	1/4	None	(10)	--	--	--	--	Liquid	None	FSAR 5.3.3.7	--	--	None	--	--	--	None
Fire Protection - Containment Hose Reel Stations	13	D	4	Check	Auto-Trip	Closed	Closed	Closed	Closed	Liquid	CIA	56	Check	15	Check (Weight Loaded)	TV (Globe)	Check	CA	None
Fire Protection - Containment Cable Penetrations	31	D	4	Check	Auto-Trip	Closed	Closed	Closed	Closed	Liquid	CIA	56	Check	15	Check (Weight Loaded)	TV (Globe)	Check	CA	None
Fire Protection - Containment RHR Platform	32	D	3	Check	Auto-Trip	Closed	Closed	Closed	Closed	Liquid	CIA	56	Check	15	Check (Weight Loaded)	TV (Globe)	Check	CA	None

Notes:

(1) Sections 7.3 and Tables 7.3-1 and 7.3-2 describe actuation signals.

(2) 1971 General Design Criterion met is listed with appropriate subparagraph in parentheses. Exceptions to General Design Criterion arrangements are noted by explanatory Updated FSAR section numbers.

(3) MOV = Electric Motor Operated Valve
TV = Trip Valve

(4) For electric motor-operated valves (MOVs) the appropriate motor control center is listed. Those which have an even number receive power from Emergency Generator 2. Those which have an odd number receive power from Emergency Generator 1 (Section 8.5). For air-operated trip valves (TVs), the power supply is designated CA for compressed air (Section 9.8). For solenoid operated trip valves (TVs), the power supply is designated SOV for solenoid operator. Secondary modes of operation are provided for MOVs in the form of manually operated (handwheels) overrides; for TVs, CA pressure can be bled off or solenoid operator can be de-energized to trip valves to fail-safe positions. Manual and check valves have no specific secondary modes of operation.

(5) Closure times listed for air-operated trip valves are calculated assuming no plug weight or stem friction. All times listed are design values.

(6) Opened to shift low head safety injection from refueling water storage tank to containment sump (Section 6.3).

(7) Containment purge isolation valves are normally shut, remote manually operated, administratively controlled valves, meeting General Design Criterion 56-(1). During refueling operations, these valves are opened and ~~will~~ trip shut automatically on receipt of a containment high-high radiation signal (Section 11.3).

may be configured to

(8) Each of the containment leakage monitoring and containment wide range pressure monitoring lines have a one-eighth inch orifice on the inside of containment located adjacent to the three-eighth inch penetration, consistent with the requirement of Safety Guide 11 (Section 1.3). Penetrations 57-3 and 57-4 have been capped inside and outside of containment.

(9) CIA can be overridden for sample inlet valves (Section 9.6.1.1).

(10) Hydraulic isolation device.

(11) Penetrations 55-2, 57-1, 57-2, and 97-3 are isolated outside of containment downstream of the containment pressure instrumentation by a sealed manual valve and pipe cap.

(12) Main Steam Isolation Valve fails closed on loss of air and fails AS-IS on loss of 125 VDC.

(13) Shutdown position in Modes 3 and 4.

(14) DBA position is for LBLOCA.

(15) Failure position is for power failure.

(16) Relief valves listed in Licensing Requirements Manual Table 5.1-1.

(17) Relief valve between isolation valves.

A nominal flow of 2,000 cfm is the design basis for the separate iodine filtration system with impregnated charcoal cells sized for 500 cfm and minimum charcoal composite efficiency of 95 percent based on the influent iodine composed of 90 percent elemental and 10 percent organic iodines.

may be

Purge exhaust is designed in the normal mode for a one air change flow rate of 30,000 cfm with connections to supplemental leak collection and release filter banks and duct system dampers arranged to reduce the concentration of any possible airborne radioactivity to levels acceptable for atmospheric discharge at an elevated release point meeting the requirements of 10 CFR 20 and 10 CFR 50. During containment refueling activities, the ~~air flow rate is 7,500 cfm and the~~ purge exhaust ~~is~~ lined up to the supplementary leak collection and release system through seismically designed ducts and dampers. The purge exhaust system is also designed with radiological monitoring to meet the requirements of 10 CFR 20 and 10 CFR 50. The purge exhaust is designed with the capability of diverting a reduced flow rate of 1,000 cfm to the process vent filters and gaseous waste blowers discharging at the top of the cooling tower in the event of a high degree of activity.

The common ventilation vent, located on top of the auxiliary building is designed to discharge low level and nonradioactive purge exhaust air at a point 10 ft higher than the turbine building roof.

Purge supply is designed for a flow rate of 27,000 cfm which is less than the normal exhaust rate to maintain a slight negative pressure in the containment. During containment refueling, air infiltrates into the containment through the purge supply system.

Reactor coolant pump motors do not constitute a major heat source since they are cooled by self-contained water coils fixed to the motor housings.

5.4.1.2 Description

Bulk air cooling of the containment is achieved by three air recirculation cooling systems with the recirculated air being cooled on passing through chilled water or river water coil banks. Cooled air is circulated by three 50 percent design capacity fans, each with a capacity of 150,000 cfm, discharging into common ductwork supplying the ventilated spaces. Air leaving the ventilated spaces is recirculated back to the supply fans via the annular space between the crane wall and containment outside wall. All three fans are normally operable; however, bulk air temperature is limited to a design maximum of 105°F with any two fans in operation. Gravity actuated back draft dampers are installed at the discharge of each fan to prevent reverse flow through an idle fan.

filters are effective for iodine removal and the pleated glass fiber type HEPA filters, at a minimum efficiency of 99.97 percent, remove particulates and charcoal fines. Each HEPA filter cell is rated at 1,000 cfm and each charcoal filter cell is rated at 500 cfm. The charcoal filter cell is of flat parallel tray design containing approximately 44 pounds of charcoal in two 2 inch beds having a total of about 8 square feet of face area. The media is new impregnated activated coconut shell charcoal, North American Carbon type G615. The media particle size is from 8 to 16 U.S. Sieve mesh with not more than 5 percent retained on 8 mesh nor more than 5 percent passing through 16 mesh. A dual fan-filter system is used, so that one may decay to a safe level before replacement with the second unit available as needed.

During shutdown periods, containment purging ventilation is provided by an exhaust system and a supply system designed to maintain a slight negative containment pressure, with provisions to handle variable flow rates up to one air change per hour, or 30,000 cfm. During unit shutdown, these systems are manually actuated if activity levels within the containment are high enough to require purging before personnel entry. This system also functions as a heating and ventilation system during periods of maintenance.

may be may be

During refueling, the purge system exhaust duct ~~is~~ lined up to the supplementary leak collection and release system. The air ~~is~~ exhausted through the filters ~~at a rate of 7,500 cfm~~ with make up air infiltrating into the containment.

The purge system exhaust circuit is provided with connections to dual filter banks consisting of prefilters, particulate-filters and charcoal filters to ensure that any radioactivity is within allowable offsite limits, as required in 10CFR20 and 10CFR50.

This circuit consists of a valved outlet from the containment, with ductwork extending to the filter banks and connected to the supplementary leak collection and release system. Filter characteristics and the supplementary leak collection and release system are described in Section 6.6. The purge exhaust air bypasses the filters and discharges to the monitored common ventilation vent when filtration is not required. Under conditions of high activity in the containment, a small quantity, 1,000 cfm of the purge exhaust air is also capable of being diverted to the process vent filters and gaseous waste blowers for discharge at the top of the cooling tower. The gaseous waste disposal system is described in Section 11.2.3. In normal unit operation, the purge exhaust and refueling circuit is inoperative and the butterfly isolation valves at the containment barrier are closed. Ventilation vent instrumentation, including radiation monitors and flowmeters, and containment radiation monitors

The ESF systems provided for satisfying these functions are:

1. A safety injection system, which injects borated water into the reactor coolant loops (Section 6.3).

2. Two separate low pressure safety injection subsystems which provide long term reactor core decay heat removal (Section 6.3).

3. The separate subsystems of the containment depressurization system (the quench spray subsystems and the recirculation spray subsystems) which, when operating together, reduce the containment temperature, return the containment to subatmospheric pressure and remove heat from the containment. The recirculation spray subsystems maintain the containment subatmospheric and transfer the heat from the containment to the river water system. The addition of a chemical (NaOH) to the quench spray reduces the concentration of airborne fission products in the containment (Section 6.4).

4. The supplementary leak collection and release system, which normally exhausts the structures contiguous to the containment (except the main steam valve cubicle), the fuel building and the waste gas shortage area to an elevated release point. Following a DBA ~~or a fuel handling accident~~ or a waste gas storage accident, the exhaust from these areas is diverted to a filter prior to being discharged at the elevated release point (Section 6.6).

5. The post DBA hydrogen control system, which, after a DBA, circulates the containment atmosphere through a thermal recombiner in which the hydrogen and oxygen are combined to form water. This system is not required for 12 hours following a DBA (Section 6.5).

A composite schematic of the ESF systems is shown in Figure 6.1-1.

The safety injection system provides for the injecting of borated water to the RCS from the accumulators following a LOCA. The three accumulators are self contained and are designed to supply water as soon as the RCS pressure drops below accumulator pressure. Continued makeup is provided by the charging pumps and the low head safety injection pumps. Both the charging and low head safety injection pumps are located outside the containment and are electric motor-driven, capable of being energized and operated rapidly and powered by the emergency buses. The pumps also ensure an adequate

6.6 SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

6.6.1 Design Bases

The supplementary leak collection and release system is designed according to the following criteria:

1. The maintenance of 0.125 inch water gauge negative pressure in most areas contiguous to the containment (with the exception of the main steam valve area, the steam generator blowdown room, the purge air duct area, east cable vault and west cable vault), containment during refueling, the waste gas storage area and the fuel building.

2. The filtration by impregnated charcoal of contaminated air for radioactive iodine removal with individual charcoal cells sized for 600 cfm.

3. The discharge of exhaust air at the SLCRS Vent.

4. The use of redundant filter banks and exhaust fans with the fans operable on emergency power.

5. Equipment and system capability of withstanding the design basis earthquake without loss of function.

6. Removal of heat from areas containing safety related equipment following a design basis accident with loss of offsite power.

6.6.2 Description

The elements of the supplementary leak collection and release system are shown in Figure 6.6-1. The design data is provided in Table 6.6-1.

The primary function of the system is to ensure that radioactive leakage from the primary containment following a DBA, ~~or radioactive release due to a fuel handling accident,~~ or radioactive material released in the waste gas storage area is collected and filtered for iodine removal prior to discharge to the atmosphere at the SLCRS Vent.

An important secondary function of the system is the removal of heat from areas contiguous to containment where equipment important to safety is located (i.e., the charging pump cubicles, the low head safety injection pump cubicles, the recirculation spray pump cubicles, the auxiliary feedwater pump room, the east and west cable vaults, the MCC room, and the safeguards pipe tunnel). Following a loss of offsite power, SLCRS is the only available means of assuring that components in these areas do not exceed the design temperatures. Temperature switches are provided to open SLCRS dampers when area temperatures exceed 110°F.

and HEPA filter is rated at 1,200 cfm. It is of flat parallel tray design, containing approximately 44 lb of charcoal in two 2 inch beds having a total of about 8 sq ft of face area. The media is new impregnated activated coconut shell charcoal. The media particle size is from 8 to 16 U. S. Sieve mesh with not more than 5.0 percent retained on 8 mesh nor more than 5.0 percent passing through 16 mesh. Charcoal cells are leak tested in accordance with Technical Specifications.

Charcoal filter efficiency is based on the attenuation of radioactive iodine with a minimum composite efficiency of 95 percent with the influent iodine composed of 90 percent elemental and 10 percent organic iodine.

A heat detection and alarm and automatically actuated water spray deluge system is provided for the charcoal cells to maintain cooling of the media and prevent ignition in the event of decay heat buildup.

The leak collection exhaust fans discharge through a duct to the SLCRS Vent. The SLCRS Vent is located on the top of the containment structure. The containment extends 144 ft above grade. The duct and supporting structure is designed to accommodate seismic forces. The release above the containment is a high-velocity discharge at an elevation of 150 ft above grade.

During refueling, the fuel building is ~~and containment are~~ normally maintained at negative pressures of 0.125 inches of water column, by the operating leak collection exhaust fan.

During Mode 6, SLCRS ventilation damper VS-D-4-18A may be ~~is normally~~ closed and the upstream containment contiguous areas ~~are~~ may not be ventilated.

If there should be a fuel building high-high radiation signal, the exhaust from the fuel building along with the rest of the leak collection system exhaust, ~~is~~ may be diverted to one of the main filter banks in the leak collection release system before being released at the SLCRS Vent.

6.6.3 Evaluation

The supplementary leak collection and release system incorporates redundant 100 percent capacity leak collection exhaust fans and main filter banks. In addition, there are redundant dampers where required. The redundant fans and dampers are connected to different emergency buses, which are capable of being supplied either from normal or emergency power sources. Thus, there is sufficient redundancy in the system to ensure system reliability. Proper operation of the system is further ensured by the capability for testing the system periodically.

The instrumentation, control and electrical equipment of the supplemental leak collection system are in accordance with IEEE 279[1]-1971 and IEEE 308[2]-1971 with one exception.

IEEE 279-1971 requires that the "protection system shall, with precision and reliability, automatically initiate appropriate protective action whenever a condition monitored by the system reaches the preset level". It also requires that this be accomplished with a single failure.

The exhausts from the contiguous areas, (with the exception of the main steam valve cubicle), the fuel building and the waste gas storage area are each monitored by a radiation monitor which automatically diverts the exhaust through a filter path when the radiation reaches a preset level. Should any of these systems fail, a redundant backup radiation monitor, located in the SLCRS Vent duct, would indicate and alarm to the operator that the preset level had been reached, which would alert the operator to manually divert the exhaust through the filters. The automatic and manual systems are redundant and on separate power supplies.

The radioactivity release from the contiguous and waste gas storage areas is expected to be much less than specified by the guidelines described in 10 CFR 100.

The supplementary leak collection and release system collects, filters and releases at an elevated point (SLCRS Vent) the leakage from the containment following a DBA ~~and refueling accident.~~ Essentially, all the leakage from the containment following a DBA flows into those containment contiguous areas which house the various containment penetrations and the engineered safeguards equipment circulating radioactive water.

Although a negative pressure of .125 inch water gauge in the areas contiguous to Containment is adequate to prevent exfiltration under DBA conditions, no credit has been taken for the collection of containment leakage through any electrical penetration in the LOCA analysis of Section 14.3. Such leakage is modeled as the release of .1% volume per day directly to the environment without regard to the specific location of the containment isolation failure.

The only credit taken for SLCRS is for the collection and filtration of leakage from ESF piping systems and components that recirculate containment sump water outside the containment. A negative pressure of .125 inch water gauge is required only for those areas.

The SLCRS Vent in the supplementary leak collection and release system is located above the top of the containment. The high velocity, coupled with the mass flow rate, ensures adequate stack effect for wind speeds below 2 m/second. At wind velocities considerably in excess of 2 m/second, some entrainment of the likewise limit the maximum lift of a fuel assembly to within 8 ft-4 inches of the normal water level in the spent fuel pit.

The fuel pool elevator, located within the spent fuel pool, utilizes push button switches. An elevator operator must hold down the up push button switch to raise the fuel pool elevator. This ensures that the operator will be able to immediately stop the elevator upon any indication of trouble or increasing radiation levels in the fuel building. The new fuel elevator may be used for repair of irradiated fuel assemblies if a mechanical stop is added to prevent raising the irradiated fuel assemblies too close to the surface.

9.12.5.2 Malfunction Analysis

137

The analysis presented in Section 14.2.1 assumes damage to ~~the entire group of~~ fuel rods in an assembly as the basis for a fuel handling accident.

9.12.5.3 Seismic Considerations

The maximum design stress for the structures and for all parts involved in gripping, supporting, or hoisting the fuel assemblies is 1/5 ultimate strength of the material. This requirement applies to normal working load and emergency pullout loads, when specified, but not to earthquake loading. To resist design basis earthquake forces, the equipment is designed so that the stress in any load bearing part is less than 0.9 times the ultimate strength for a combination of normal working load plus design basis earthquake forces.

9.12.6 Tests and Inspections

Prior to initial fueling, preoperational checkouts of the fuel handling equipment were performed to ensure proper performance of the fuel handling equipment and to familiarize station operating personnel with operation of the equipment. A dummy fuel assembly was utilized for this purpose.

Upon completion of core loading and installation of the reactor vessel head, certain mechanical and electrical tests are performed prior to criticality. The electrical wiring for the rod drive circuits, the rod position indicators, the reactor trip circuits, the incore thermocouples, and the reactor vessel head water temperature thermocouples are checked at the time of installation. The checks are repeated on these electrical items before unit operation.

The precritical tests are performed in close cooperation with the Station Instrument Department, the Station Electrical Engineer, and the Station Test Section.

the normally radioactive process auxiliary systems are at acceptable levels.

Normally nonradioactive systems which may become contaminated by leaks from radioactive systems will be monitored continuously to ensure that no condition hazardous to the operators or to the general public will be developed.

Selected process and effluent radiological monitoring channels will be designed as Seismic Category I to ensure their availability under accident conditions.

In the event of an accident, the process and effluent radiological monitoring system, in conjunction with the area radiation monitoring system (Section 11.3.4) will provide information on the concentration and dispersion of radioactivity throughout the unit. This will enable operating personnel to evaluate the severity and mitigate the consequences of an accident.

The following automatic actions will be initiated by the process and effluent radiological monitoring system to mitigate the consequences of postulated accidents:

1. Containment refueling ventilation will be terminated in the event of a refueling accident inside containment. Any activity released prior to accomplishing containment isolation ~~will~~ may be directed to the Supplementary Leak Collection and Release System (SLCRS) filters.

2. Fuel building ventilation will be through the Supplementary Leak Collection and Release System (SLCRS). During fuel handling, flow ~~is~~ may be directed through the charcoal filters.

3. Ventilation flow from all areas enclosing radioactive fluids, including loss-of-coolant accident recirculation fluids, will be diverted through charcoal filters in the event of a release of activity into that area.

4. All effluent discharges from the secondary system will be terminated in the event of a steam generator tube rupture.

5. All effluent flow to the environment will be automatically terminated or processed in the event of a high-high activity alarm condition or an ESF signal, depending on the nature of the initiating event.

11.3.3.2 Continuous Monitoring

The monitoring system consists of separate independent channels each having:

1. Radiation detector with remotely operated check source except for steam effluent monitors: SGADV, MSSV and AFTEX monitors

All equipment is identical with that of the two gaseous waste channels. Normally, the sample is composed of effluent from the leak collection areas, fuel building, and waste gas storage tank area which are each individually monitored. A high-high activity alarm signal from any of these sources will cause all of these effluents to pass through a prefilter/charcoal/HEPA filter complex in the supplementary leak collection and release system before discharge via the Reactor Building/SLCRS release point.

Two high range noble gas monitoring systems continuously draw a sample from the Reactor Building/SLCRS exhaust line upstream of the point of discharge. These systems are operationally and functionally similar to the Gaseous Waste high range noble gas monitors discussed in Section 11.3.3.3.23.

11.3.3.3.5 Auxiliary Building Ventilation Exhaust Monitors

The auxiliary building ventilation system consists of exhaust system "A" and exhaust system "B." A continuous gas sample from each exhaust system is first passed through an in-line, easily removable, charcoal filter cartridge (as desired) and then analyzed by a gas monitor similar to the gas sampler/monitor equipment described for the gaseous waste monitoring system. The building exhaust is normally discharged to the atmosphere via the ventilation vent duct. A high-high alarm will cause the exhaust effluent to be diverted through a prefilter/charcoal/HEPA filter complex in the supplementary leak collection and release system before atmospheric discharge via the elevated release point. With manual damper VS-D-7-8A normally closed during Mode 6, operator action is required to manually open the damper to accomplish diversion in the event of a high-high alarm.

The charging pump cubicle exhaust is not monitored by an auxiliary building ventilation monitor. The cubicle exhaust is discharged through the supplementary leak collection and release system release point, which is monitored by an effluent radiation monitor. A CIA signal will cause the charging pump cubicle exhaust effluent to be diverted through a prefilter/charcoal/HEPA filter complex in the supplementary leak collection and release system before atmospheric discharge.

11.3.3.3.6 Fuel Building Ventilation Exhaust Monitors

The fuel building ventilation exhaust is monitored by two redundant in-line detectors. A high-high radiation alarm ~~will~~ may automatically divert the flow through the prefilter/charcoal/HEPA filter complex of the supplementary leak collection and release system before discharge to the elevated release duct. It will also activate the fuel building and containment evacuation alarm.

11.3.3.3.7 Containment Purge Exhaust Monitor

The containment purge exhaust is monitored by two redundant inline detectors. During the initial containment purge (prior to allowing personnel entry into containment for refueling normal

operations) an activity alarm will signal the operator to manually actuate damper valves to divert the purge exhaust through the prefilter/charcoal/HEPA filter complex for subsequent discharge via the elevated release point. For the likely no-radiation-alarm condition, the normal purge cycle is completed and the containment is opened to atmosphere permitting entry for refueling operations. During refueling, the purge/exhaust system may maintain~~s~~ the containment at a slightly negative pressure. ~~in accordance with the supplementary leak collection and release system (SLCRS) design as stated in UFSAR Section 6.6.~~ Flow ~~will~~, may ~~at all times during refueling,~~ be directed through the SLCRS prefilter/charcoal/HEPA filter complex. A high-high activity alarm from either of the containment purge exhaust monitors, ~~will~~ may automatically close the purge supply and exhaust isolation dampers in the containment building and activate the fuel building and containment evacuation alarm. The airborne activity in the containment atmosphere ~~is~~ may be subsequently discharged at a controlled rate through the prefilter/charcoal/HEPA filter complex in the supplementary leak collection and release system to the elevated release point above the containment building.

11.3.3.3.8 Leak Collection Area Gas Monitor

The leak collection area exhaust is monitored by analyzing a sample continuously drawn from the discharge flow stream. The sample is passed through a removable charcoal filter cartridge (as desired) and then drawn into a fixed lead-shielded sampler enclosing the detector. The sample activity is measured and then returned to the leak collection area exhaust discharge line. A high-high activity alarm signal will cause automatic diversion of the air flow through the prefilter/charcoal/HEPA filter complex to the elevated release point above the containment. A purge system is integral with the gas monitoring system for flushing the sampler with clean air for purposes of calibration.

11.3.3.3.9 Waste Gas Tank Vault Ventilation Gas Monitor

The ventilation exhaust from the waste gas tank vaults is monitored by analyzing a sample continuously drawn from the ventilation discharge flow stream. This monitoring system and its operation are identical with the monitoring system and operation described for the leak collection area gas monitor.

11.3.3.3.10 Component Cooling Water Monitor

The component cooling water monitor continuously analyzes a composite sample drawn from the downstream leg of the primary plant component cooling water heat exchangers. The sample is drawn through an off-line sampler system containing a detector inserted in a well located in the flow stream of the off-line sampler. Detection of radioactivity indicates leakage of radioactive effluent from the reactor coolant system or an auxiliary system into the component cooling water system.

TABLE 11.3-7

POSTULATED CONTROL ROOM ACCIDENT DOSE, REM(5)(6)

(Design Basis Accidents at Unit 1)

Accident	Thyroid	Gamma	Beta	Notes
Main Steam Line Break				
Co-incident Spike	26.0	2.9E-3	3.7E-2	2
Pre-incident Spike	29.0	1.5E-3	2.0E-2	2
Small Line Break	27.0	3.0E-3	2.6E-2	4
Steam Generator Tube Rupture				
Co-incident Spike	3.13	8.12E-4	2.22E-2	4
Pre-incident Spike	8.65	9.30E-4	2.34E-2	4
Rod Ejection Accident	12.0	1.1E-3	8.9E-3	4
~~Fuel Handling Accident~~	~~4.3~~	~~1.4E-2~~	~~8.3E-1~~	~~4~~
Locked Rotor Accident	9.69	7.35E-2	1.16	1
Loss of Auxiliary AC Power	8.0E-1	7.1E-5	2.7E-3	4
Waste Gas System Rupture				
Line Break	--	< 1.0E-2	< 1.0	4
Tank Rupture	--	7.1E-4	1.3	4
DBA LOCA	14.3	1.7E-1	4.0E-1	3
		TEDE		
Fuel Handling Accident		2.2E+00		7

NOTES:

1. Control Room isolation by area radiation monitor signal based on a setpoint with a safety limit dose rate of 1 mrem/hr gamma in the Control Room.
2. Isolation by manual operator action at T=30 minutes post-accident. In support of Alternate Repair Criteria for steam generators (ref. USNRC GL 95-05) the MSLB thyroid doses were maximized within applicable limits in order to establish the maximum allowable accident-induced leakage against which tube leakage projections, based on voltage indication, are compared. Current values are based on 8.0 gpm primary-to-secondary leakage (0-2 hour Exclusion Area Boundary thyroid dose limiting). See Section 14.2.5.1.3.
3. Control Isolation actuated by CIB signal.
4. No action required.
5. References: ERS-SFL-93-005 r0, ERS-SFL-92-033 r1, 12241/14110.39-UR(B)-456, 14110.39-UR(B)-457 r0, ERS-SFL-89-021 r1, ERS-SFL-95-008 r2.
6. Listed dose values represent the limiting bounding value.
7. Minimum 16,900 cfm purge for 30 minute duration after release to environment ends.

14.2 STANDBY SAFEGUARDS ANALYSIS

14.2.1 Fuel Handling Accident

This section discusses the design features which preclude a serious accident during fuel handling, and then verifies that the consequences of the worst case assumptions meet ~~10CFR100~~ guidelines. 10CFR 50.67 limits and NUREG-0800 (SRP 15.0.1)

The following representative fuel handling accidents are evaluated during the course of design to ensure that no hazards are created:

1. A fuel assembly becomes stuck inside the reactor vessel, in the penetration valve in the transfer carriage, or the transfer carriage itself becomes stuck. In this case, the criterion is to ensure cooling of the fuel.

2. A fuel assembly is dropped onto the floor (or onto other assemblies which may be present) of the refueling cavity or spent fuel pool. In this case, assuming that 137 ~~all~~ fuel rods ~~in the dropped assembly~~ are ruptured, the criterion is to ensure that the offsite dose is ~~within 10CFR100 guidelines.~~ acceptable as described above.

14.2.1.1 Accident Description

The possibility of a fuel handling incident is very remote because of the many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety.

The fuel handling manipulators and hoists are designed so that fuel cannot be raised above a position which provides adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pool area.

In the spent fuel pool, the design of storage racks and manipulation facilities is such that:

1. Fuel at rest is positioned by positive restraints in an oversafe ($k_{eff} \leq 0.95$), always subcritical, geometrical array, with no credit taken for boric acid in the water.

2. Only one fuel assembly can be manipulated at a time.

3. A minimum boron concentration of 400 ppm will maintain $k_{eff} \leq 0.95$ for fuel assembly misplacement. Misplacement of a new fuel assembly of 5.0 weight percent enrichment in a Region 2/Region 3 location, surrounded by locations filled with fuel of the highest permissible reactivity, could cause k_{eff} to exceed 0.95 without soluble boron.

friction force. This would absorb the shock and thus limit the force on the individual fuel rods.

After the reactor is shutdown, the fuel rods contract during the subsequent cooldown and would not be in contact with the bottom plate of the assembly.

Considerable deformation would have to occur before the rod would make contact with the top plate and apply any appreciable load on the fuel rod. Based on the above, it is unlikely that any damage would occur to the individual fuel rods during handling. If one assembly is lowered on top of another, no damage to the fuel rods would occur that would breech the integrity of the cladding.

All these safety features make the probability of a fuel handling accident very low. The worst case, which is hypothesized with respect to the release of fission products to the environment, is a fuel handling accident in the fuel handling building.

14.2.1.2 ~~Method of Analysis~~ <INSERT>

The fuel handling accident is classified as an ANS Condition IV event; i.e., faults that are not expected to occur but are postulated because their consequences include the potential for the release of significant amounts of radioactive material.

The fuel handling accident is postulated to occur in the fuel building and in the containment. Environmental release from the containment is precluded by a design which automatically isolates the containment following the detection of radioactivity by the redundant containment purge monitors (Section 11.3.3).

The fuel handling accident sequence of events consists of the dropping of one fuel assembly on another fuel assembly in the fuel pool, resulting in cladding damage to the fuel rods in the dropped assembly plus additional rods in the struck assembly with subsequent instantaneous release of all the gap radionuclide inventory.

The gap radionuclide inventory is based on the minimum time after refueling shutdown of 150 hours and peak inventories for the damaged fuel assemblies. The fuel pool water provides retention capabilities for radioiodines as described in Table 14.2-6.

The radioactivity released from the fuel pool into the fuel building atmosphere is filtered by the supplementary leak collection and release system (Section 6.6).

The radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building (Section 9.12) meet the requirements of 1971 GDC 61.

The assumptions applied to the evaluation of the release of radioactivity from the fuel and the fuel building are provided in

Table 14.2-6. Gap inventories of fission products were determined using the release fractions identified in Regulatory Guide 1.25 except for Iodine-131, for which the release fraction is increased 20 percent in accordance with NUREG/CR-5009.

14.2.1.3 Fission Product Inventories

Core specific inventories (Curies per metric ton of uranium) of fission products were estimated with the ORIGEN-2 code.

The results of the ORIGEN calculations for isotopes that contribute to the thyroid and whole-body doses are given in Table 14.2-6A, while Table 14.2-6B lists pertinent data for the isotopes of interest. Data and assumptions used in the dose calculations are given in Table 14.2-6.

14.2.1.4 Offsite Exposure from Accident During Refueling

During refueling, the fuel building will be maintained at a minimum negative pressure of 0.125 inches W.G., with a maximum exhaust flow rate of 3,000 cfm. The maximum fuel building exhaust flow rate of 3,000 cfm was assumed in the fuel handling accident of the Control Room Dose Analysis.

During refueling, the containment is maintained by the supplementary leak collection and release system at a negative pressure of 0.125 inches W.G.

Exhaust flow from the fuel building will discharge through the main supplementary leak collection and release system filter bank to the SLCRS Vent.

The doses at the Beaver Valley EAB from the specified fuel handling accident are tabulated below. The doses are based on the release of all gaseous fission product activity in the gaps of 298 fuel rods in highest-power assemblies.

Thyroid dose, rad	=	14.60
Whole-body dose, rem		
Beta dose, D_β	=	2.96
Gamma dose, D_γ	=	0.95
Whole-body total, rem	=	3.91

These potential doses are well within the exposure guideline values of 10 CFR 100, paragraph 11. As defined in Standard Review Plan 15.7.4, <u>Radiological Consequences of Fuel Handling Accidents</u>, "well within" means 25% or less of the 10 CFR 100 guidelines, or values of 75 rad for thyroid doses and 6.25 rem for whole-body doses.

The doses for the Beaver Valley Common Control Room were also analyzed for the specified fuel handling accident. The doses for the duration of the accident are:

Thyroid dose, rem	=	4.33
Whole-body dose, rem		
Beta dose, D_β	=	0.83
Gamma dose, D_γ	=	0.014

The doses are within the criteria of 10 CFR 50, Appendix A, General Design Criterion 19 and are acceptable. Table 14.2-6 tabulates significant analysis parameters. Table 11.3-7 tabulates results.

In the event of a fuel handling accident within the containment structure, a high-high radiation level in the purge duct discharge will trip the purge line isolation valves and isolate the containment from the supplementary leak collection and release system. During refueling in the containment, the containment isolation valves of the containment purge supply and exhaust systems are not required to close since the containment purge exhaust duct is lined up to the seismically supported leak collection system and filter train. No credit is taken for containment isolation in the safety analysis for a fuel handling accident in the containment. Since the containment has been maintained at subatmospheric pressure, there will be no driving force for discharge of gas from the containment, which will result in no release of activity to the environment. Once the nature of the activity release within the containment has been ascertained, a containment ventilation operation may be manually initiated, the containment iodine removal filter may be placed into operation, or other corrective action taken. For analysis purposes no credit is taken for containment isolation. For these conditions, the site dose would be the same as that obtained for a fuel handling accident in the fuel building.

14.2.2 Accidental Release of Waste Liquid

14.2.2.1 Identification of Causes and Accident Description

Accidents in the auxiliary system which could result in the release of waste liquid may involve the rupture or leaking of various components.

14.2.2.2 Analysis of Effects and Consequences

Liquid processing components are located within the auxiliary building, and any liquid leakage or release from the components is locally collected and transferred to sumps for subsequent pumping into the liquid waste disposal system.

Curbs and floor drains to the sump are employed to minimize the effect of leakage and spills. Outboard seal leakage from the charging pumps (Section 9.1) is contained in this manner. The ventilation system collects any gaseous radioactivity and discharges it to the monitored ventilation vent as discussed in Section 9.13.

<INSERT Page 14.2-3>

14.2.1.2 Radiological Consequence Analysis Methods, Assumptions and Results

The fuel handing accident is classified as an ANS Condition IV event; i.e., faults that are not expected to occur but are postulated because their consequences include the potential for the release of significant amounts of radioactive material.

This DBA is described in NRC Regulatory Guide 1.183 and NUREG-0800 Chapter 15, Section 15.0.1. The parameters and assumptions used to perform the radiological consequence analysis are summarized in Tables 14.2.6 and 14.2.6a. The dose calculation methodology is provided in Appendix 14B.

The accident occurs while moving a fuel assembly in either the fuel building fuel storage pool or in the reactor building containment cavity or transfer canal. The assembly is dropped, resulting in rupture of 137 fuel rods (in the dropped assembly plus other assemblies that may be struck) and release of radioactive iodine and noble gas into the pool water. The extent of damage has been determined by performing an analysis using the limiting drop conditions and considering the weight of the 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 assemblies has been considered.

As the iodine percolates to the pool surface much of it will be "scrubbed out" and remain in the water. This reduces the amount of iodine that is released from the damaged rods by a factor of 200 before it becomes airborne within the building. The noble gas activity released from the damaged rods is all released to the building with no removal by pool water scrubbing. After becoming airborne, the radioactivity is released to the environment assuming a constant air flow rate (exponential activity removal rate). The analysis model uses a conservatively calculated release rate constant that results in 99.9999% of the activity being released to the environment in the two hours immediately following the accident. Because the accident conditions may include having any of the reactor building containment penetrations open (including the equipment hatch or personnel airlock), and the release may be via any one or a combination of penetrations, the most restrictive release point atmospheric dispersion factor(s) are conservatively applied to the entire release. For the radiological consequence analysis to remain valid, the radioactivity release must be via one of these three points. Additionally, the analysis conservatively does not take credit for any pre-release filtration or iodine plate-out.

<INSERT Page 14.2-3 continued>

The environmental radioactivity release is transported to the dose receptor points assuming dispersion due to diffusion and local meteorological conditions with no consideration given to radioactive decay during transport or to gravitational settling. For control room personnel, protective systems are provided to mitigate the consequences of a fuel handling accident. The capability to purge the control room of radioactivity after the environmental release has ended is the only protective system feature that is considered in the dose analysis. The minimum system flow rate and operation time used in the analysis is provided in Table 14.2.6.

The dose analysis results are:

	TEDE (rem)
Control Room	Table 11.3-7
EAB	1.7E+00
LPZ	9.5E-02

These postulated fuel handling accident doses are within the limits provided in 10 CFR 50.67 of 25 rem TEDE for the EAB and LPZ, and 5 rem TEDE for the control room. Additionally the accident doses are within the more restrictive criteria provided in NUREG-0800 Section 15.0.1 of 6.3 rem TEDE for the EAB and LPZ.

References to Section 14.2 (CONT'D)

12. S. Altomare and R. F. Barry, "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7758, Westinghouse Electric Corporation (September 1971).

13. R. F. Barry, "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269-26, Westinghouse Electric Corporation (September 1963).

14. "RADIOISOTOPE, A Computer Program for Calculating Residual Activities in a Closed System After One or More Decay Periods," RP-1, Stone & Webster Engineering Corporation (November 1972).

15. "ACTIVITY, A Computer Program for Calculating Fission Product Activity in Fuel, Coolant, and Selected Tanks for a Nuclear Power Plant," RP-3, Stone & Webster Engineering Corporation (January 1973).

16. "IONEXCHANGER, A Computer Program for Determining Gamma Activities in Ion Exchangers or Tanks as a Function of Time for Constant Feed Activity," RP-2, Stone & Webster Engineering Corporation (December 1972).

17. M. S. Baldwin, M. M. Merrian, H. S. Schenkle, and D. J. Van De Walle, "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," WCAP-8424, Revision 1, June 1975.

18. Calculation ERS-SFL-89-021, Safety Analysis of the Dose Consequences of a Locked Rotor Accident at BVPS-1 with 18% Fuel Failure -- EAB, LPZ, Control Room.

19. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

20. Combined BV1-BV2 Control Room Habitability Due to Design Basis Accidents (except LOCA) at BV1, Calculation 12241/ 14110.39-UR(B)-456-0, 1987.

21. Doses to the BV1 Control Room Due to LOCA at BV1, Calculation 14110.39-UR(B)-457-0, 1987.

~~22. Calculation ERS-SFL-92-025 r0, Safety Analysis of the Dose Consequences of a Fuel Handling Accident at BVPS-1 to the Common Control Room, 1992.~~

23. Calculation ERS-SFL-92-033 r1, Combined Control Room Doses Due to SGTR at Unit 1, 1996.

24. Calculation ERS-SFL-93-005, Safety Analysis of Consequences of Control Room Damper Response Delay (Limitorque 10 CFR 21) Unit 1 Accidents.

<INSERT Page 14.2-57>

22. BVPS Calculation ERS-JTL-99-009, Safety Analysis of the Radiological Consequences of a Fuel Handling DBA at BVPS Unit 1, Control Room, EAB and LPZ doses, 2000

References to Section 14.2 (CONT'D)

25. Calculation ERS-SFL-95-008, Safety Analysis of the Common Control Room, EAB, LPZ, Doses from a Main Steam Line Break Outside of CNMT at U1 with Increased Primary-to-Secondary Leakage.

26. USNRC, "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," Generic Letter 95-05.

27. USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." NUREG-0800.

28. USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.

TABLE 14.2-6

ASSUMPTIONS USED FOR THE FUEL HANDLING ACCIDENT RADIOLOGICAL ANALYSIS

Core power level, MW_t	~~2766~~ 2705
Fuel enrichment, weight % U*	~~5.0~~ 3.6 - 5.0
~~Fuel burnup, MWD/MTU~~	~~60,000~~
~~Specific power, kW/kgU~~	~~41.16~~
Fuel assemblies in core	157
Power peaking factor	1.65
Number of failed fuel rods	~~298~~ 137
Core inventory released to gap, %	
Iodine-131	~~12~~ 8
~~Other iodines~~	~~10~~
Krypton-85	~~30~~ 10
~~Xenon-133~~	~~10~~
Other ~~xenons~~	~~10~~ 5
~~Iodine composition, %~~	
~~Elemental~~	~~99.75~~
~~Organic~~	~~0.25~~
Pool decontamination factors (total)	
~~Elemental~~ Iodine	~~133~~ 200
~~Organic iodine~~	~~1~~
Noble gases	1
~~Filter decontamination factors~~	
~~Elemental iodine~~	~~20~~
~~Organic iodine~~	~~20~~
~~Noble gases~~	~~1~~
Atmospheric diffusion factor (χ/Q), sec/m^3	~~8.91×10^{-4}~~ Tables 2.2-11a, 2.2-11b
Breathing rate, m^3/sec	$3.\cancel{47}5 \times 10^{-4}$

Basis for core inventory

Core fraction, days at 100% power	1/3, 500
	1/3, 500
	1/3, 500
Between cycle cooldown time (days)	40

Table 14.2-6 (Cont'd.)

ASSUMPTIONS USED FOR THE FUEL HANDLING ACCIDENT ANALYSIS

~~Control Room Habitability Analysis Parameters~~

Release Duration, ~~days~~ hours	~~30~~ 2
Release rate, 1/sec**	1.92E-03 ~~0.001599~~
Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
Control Room isolation, T= min ***	30 ~~None~~
Post isolation unfiltered inleakage, cfm ***	Ø
Control Room purging, cfm	16,900 ~~None~~
Purge timing, T= min	120-150
Control Room χ/Q values, sec/m^3	Table 2.2-12
~~Analysis reference (Control Room only)~~	~~ERS-SFL-92-025 r0~~

* *represents 99.9999% of the activity in the fuel building atmosphere being released in 2 hours. This release rate is greater than that suggested by the building ventilation flow rate and volume. Analysis assumed instantaneous release from fuel to the fuel building atmosphere. ~~Release rate assumption differs from that used for offsite analysis in that the offsite analysis assumed an instantaneous release. The Control Room modeling requires a continuous intake rate.~~

* Core activity content was determined for 3.6 w/o and 5 w/o enrichments and the maximum activity for each radionuclide was selected for the analysis.

*** The control room ventilation parameters used in the analysis represent the bounding condition. If isolation timing changes, or if isolation does not occur at all, the accident consequences will be reduced. If isolation does occur and any unfiltered inleakage occurs, the accident consequences will be reduced.

TABLE 14.2-6a

RESULTS OF ORIGEN-~~2~~ CALCULATIONS FOR RADIONUCLIDES OF IODINE, KRYPTON, AND XENON AT ~~150~~ 100 HOURS COOLING TIME

<INSERT>

Radionuclide	Curies per MTU
I-131	6.519×10^5
I-132	4.149×10^5
I-133	1.473×10^4
Kr-85	1.612×10^4
Xe-131m	1.129×10^4
Xe-133	1.138×10^6
Xe-133m	1.408×10^4

<INSERT Table 14.2-6a>

Isotope	Activity in Core (Ci)
Kr-85m	3.64E+00
Kr-85	7.59E+05
Xe-131m	9.19E+05
Xe-133m	1.92E+06
Xe-133	1.03E+08
Xe-135m	6.04E+02
Xe-135	1.99E+05
I-131	5.23E+07
I-132	4.41E+07
I-133	5.45E+06
I-135	3.71E+03

TABLE 14.2-6b

<DELETE>

RADIONUCLIDE PROPERTIES USED IN THE FUEL HANDLING ACCIDENT ANALYSIS

Radionuclide	Dose Conversion, Rads/Curie	E_{β} (Mev)	E_{γ} (Mev)
Iodine-131	1.48×10^{6}	------	------
Iodine-132	5.35×10^{4}	------	------
Iodine-133	4.0×10^{5}	------	------
Krypton-85	------	0.223	0.002
Xenon-131m	------	0.140	0.164
Xenon-133	------	0.146	0.045
Xenon-133m	------	0.155	0.042

The above expression is valid for temperatures above 1,100 C. Below 1,100 C fission gas release occurs mainly by two temperature independent phenomena, recoil and knock-out, and is predicted by using D' at 1,100 C. The value used for D' (1673 K) based on data at burnups greater than 10^{19} fissions per cc, accounts for possible fission gas release by other mechanisms as well as pellet cracking during irradiation.

The diffusion coefficient for iodine isotopes was conservatively assumed to be the same as for Xe and Kr. Toner and Scott[3] observed that iodine diffuses in uranium dioxide at about the same rate as Xe and Kr and has about the same activation energy. Data reported by Belle[4] indicates that iodine diffuses at slightly slower rates than Xe and Kr.

With the diffusion coefficient determined for the fuel temperature region of interest, the fraction of radioactive fission gas which crosses the fuel boundary into the fuel rod gap is found from:

f = 3 * SQRT (D'/λ) * ((COTH (λ/D')) - (D'/λ)) (14B.2-2)

where:

f = fraction of a given radioactive fission gas in fuel rod gap

λ = fission gas decay constant, per second

D' = diffusion coefficient, per second

The above expression is the steady-state solution of the diffusion equation in spherical geometry as given by Booth[5].

Table 14B-1 lists the total core activities as well as activities present in the gap for each pertinent isotope obtained using the above equations and the fuel temperature distribution given in Table 14B 2.

The activities in the reactor coolant as well as in the volume control tank, pressurizer and waste gas decay tanks are given in Chapter 11, including the data on which the computation of these activities is based.

14B.3 FUEL HANDLING SOURCES

~~The inventory of fission products in a fuel assembly is dependent on the rating of the assembly. The parameters used for the calculations of the highest rated assembly to be discharged are summarized in Table 14B-3, while the associated activities at the time of shutdown are given in Table 14B-4.~~

~~The expected end-of-life temperature and power distributions were calculated by using the radial and axial power peaking factors of~~

<INSERT>

1.27 and 1.37, respectively. The conservative end-of-life temperature and power distributions were calculated by using the same radial power peaking factor as in the expected case, but with a higher axial power peaking factor of 1.69. Thus, the temperature/volume distribution in the fuel is changed and the maximum temperature is increased (Table 14B-3), resulting in an increased fraction of fission products in the fuel-cladding gap (Table 14B-4).

14B.4 REACTOR COOLANT FISSION PRODUCT ACTIVITIES

The parameters used in the calculation of the reactor coolant fission product inventories, together with the pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness, are summarized in Table 14B-5, while the results of the calculations are presented in Table 14B-6. In these calculations, the defective fuel rods were assumed to be present at the initial core loading and were uniformly distributed throughout the core. Thus, the fission product escape rate coefficients were based upon the average fuel temperature. The calculations were performed for the prevailing temperature upstream of the regenerative heat exchanger. The coolant density correction of 1.35 is made in order to obtain the correct activities at the downstream temperature.

The fission product activities in the reactor coolant during operation with small cladding defects (fuel rods containing pinholes or fine cracks) in 1 percent of the fuel rods were computed using the following differential equations:

For parent nuclides in the coolant:

$$\frac{dN_{wi}}{dt} = D*V_i*N_{ci} - \left[l_i + R*E_i + \frac{B'}{(B_o - t*B')} \right] *N_{wi} \qquad (14B.3\text{-}1)$$

for daughter nuclides in the coolant:

$$\frac{dN_{wj}}{dt} = D*V_j*N_{cj} - \left[l_j + R*E_j + \frac{B'}{(B_o - t*B')} \right] *N_{wj} + l_i*N_{wi} \qquad (14B.3\text{-}2)$$

where:

N = population of nuclide

D = fraction of fuel rods having defective cladding

R = purification flow, coolant system volumes per second

B_o = initial boron concentration, ppm

<INSERT Section 14B.3>

Activities in the core were calculated using the computer code ORIGEN as described in NUREG/CR-0200, and using parameter values specific to the physical and chemical makeup of the fuel and to the reactor operation. Because uranium enrichments may change from cycle to cycle and these changes may cause an increase in certain nuclides, core radionuclide inventory is calculated for a minimum expected enrichment and again for a maximum expected enrichment. The assumed core inventory used in radiological analysis is composed of a selection of the maximum value for each nuclide for the range of expected enrichments. When used in the fuel handling accident radiological analysis, this activity is also reduced to account for delay time specified in the facility Technical Specifications which limits the post criticality time duration to move fuel assemblies. Additionally, a conservative radial peaking factor of 1.65 is applied to increase the activity content to ensure that the maximum power assembly is considered in the analysis.

The postulated fuel rod gap activities are taken from Regulatory Guide 1.183. These are 0.10 Kr-85, 0.08 I-131 and 0.05 for other iodines and noble gases total activity in the core.

14B.8.5 Updated Dose Calculation Models

Commencing with analyses performed in 1995, the whole body-gamma dose, beta skin dose, and thyroid dose commitments described in Section 14B.8.1 - 14B.8.4 have been calculated using the dose quantities described in ~~this section.~~ the BVPS Unit 2 UFSAR section 15A.2.

Effective Dose Equivalent (EDE) as described in ICRP-26[11]. Replaces the traditional whole body gamma dose quantity. Like the whole body dose it replaces, the EDE model assumes that the receptor is immersed in a semi-infinite cloud. The EDE model estimates the dose to each organ in the body due to radiation from this cloud, applies a weighting factor to each organ dose, and sums the weighted doses to obtain the EDE.

$$D_{EDE} = \chi / Q \times \sum_{i} (Q_i \times C_{EDE_i})$$

where:

D_{EDE} = Effective Dose Equivalent (EDE)
Q_i = Activity of nuclide i released
χ/Q = Atmospheric dispersion factor
C_{EDE_i} = Dose conversion factor for nuclide i (DOE/EH-0070, 1988)[15]

For the control room dose analyses, the EDE is corrected to account for the finite volume of the control room using the method of Murphy-Campe[13].

$$D_{EDE_{CR}} = \chi / Q \times \frac{V^{0.338}}{1173} \times \sum_{i} (Q_i \times C_{EDE_i})$$

where:

$D_{EDE_{CR}}$ = Effective Dose Equivalent (EDE) for control room
V = volume of control room, ft^3

Skin Dose Equivalent (skin DE) as described in ICRP-26. Replaces the traditional beta skin dose quantity. Assumes that the receptor is immersed in a semi-infinite cloud.

$$D_{SKIN} = \chi / Q \times \sum_{i} (Q_i \times C_{SKIN_i})$$

where:

D_{SKIN} = Skin Dose Equivalent (skin DE)
Q_i = Activity of nuclide i released

Delete

χ/Q = Atmospheric dispersion factor

C_{SKIN_i} = Dose conversion factor for nuclide i (DOE/EH-0070, 1988)

Thyroid Committed Dose Equivalent (thyroid CDE) as described in ICRP-26 and ICRP-30[12]. Replaces the traditional thyroid dose quantity based on the critical organ model of ICRP-2[7] used in TID14844[1].

$$D_{CDE_{thy}} = \chi / Q \times \sum_i (Q_i \times C_{CDE_i} \times BR)$$

where:

$D_{CDE_{thy}}$ = Thyroid Committed Dose Equivalent (CDE)

Q_i = Activity of nuclide i released

χ/Q = Atmospheric dispersion factor

BR = Breathing rate
3.47E-4 m^3/sec, 0-8 hours
1.75E-4 m^3/sec, 8-24 hours
2.32E-4 m^3/sec, >24 hours
3.47E-4 m^3/sec, 0-30 day control room analysis

C_{CDE_i} = Dose conversion factor for nuclide i (USEPA FGR11, 1988)[16]

Delete

14B.9 CONTAINMENT LEAKAGE MODEL - DBA CASE[8]

This section describes the model used to estimate the quantity of radionuclides released to the environment by leakage from the containment building, using design basis assumptions. The realistic case leakage model is described in Section 14B.10.

14B.9.1 Radioiodine

Figure 14B-1 illustrates, schematically, the leakage model. The containment free volume is assumed to consist of two regions: a sprayed region and an unsprayed region. The processes acting simultaneously on the activity in the unsprayed region are:

a. Radioactive decay
b. Leakage from containment
c. Thermally induced exchange with sprayed region

For the sprayed region, scavenging of iodine by chemical sprays is added to the list above. This scavenging is effective on elemental and particulate species of iodine. The chemical removal continues until the maximum spray decontamination factor (DF) is reached. This DF is based on the iodine concentration in the recirculation spray and the iodine partitioning factor.

ATTACHMENT G-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 73

Applicable Draft UFSAR Changes

TABLE 1.8-1 (Cont)

RG No. 1.23, Rev. 0
UFSAR Reference Section 2.3.3

ONSITE METEOROLOGICAL PROGRAMS (FEBRUARY 17, 1972)

Onsite meteorological programs for Beaver Valley Power Station - Unit 2 will follow the guidance of this regulatory guide.

RG No. 1.24, Rev. 0
UFSAR Reference Section 15.7.1.3

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE (MARCH 23, 1972)

Beaver Valley Power Station - Unit 2 evaluation of the potential radiological consequences of a pressurized water reactor radioactive gas storage tank rupture meets the intent of this regulatory guide. The following alternatives were considered prudent:

Paragraph C.1.a

In recognition of specific plant equipment arrangements for gaseous waste handling, the system component producing the worst environmental impact was identified and additional conservatism was appropriately applied.

Paragraph C.2

Atmospheric diffusion (X/Q) values were calculated using the latest approved techniques which are provided in Regulatory Guide 1.145.

RG No. 1.25, Rev. 0
~~UFSAR Reference Section 15.4.7~~

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 (MARCH 23, 1972)

~~The assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility at Beaver Valley Power Station - Unit 2 meet the intent of this regulatory guide with the following alternative:~~

~~Paragraph C.1.j specifies iodine removal efficiencies of 90 percent for inorganic species and 70 percent for organic species. However, the efficiencies used are those given in Table 2 of Regulatory Guide 1.52 (that is, 95 percent for both~~

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

TABLE 1.8-1 (Cont)

~~organic and inorganic species if 2-inch charcoal bed depth is provided; 99 percent if 4 or more inches of charcoal bed depth is provided) since these represent more realistic values.~~

~~Paragraph C.1.d specifies that the analysis should be performed assuming 10% of the total radioactive iodine in the rods of the time of the accident. However, the iodine percentages used are 12% I-131 and 10% of the other iodine nuclides. This is in keeping with NUREG-5009, as referenced by the USNRC in the safety evaluation report for license amendment 12.~~

~~BVPS takes some alternatives to Section C.3 dose calculation methodology.~~

RG NO. 1.26, Rev. 3

UFSAR Reference Section 3.2.2

QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS (FEBRUARY 1976)

Quality group classifications and standards for water-, steam-, and radioactive-waste-containing components of Beaver Valley Power Station - Unit 2 meet the intent of Regulatory Guide 1.26 with the following alternatives:

1. The safety class terminology of ANSI N18.2 and ANSI 18.2a-1975 is used instead of the quality group terminology. Thus, the terms Safety Class 1, Safety Class 2, Safety Class 3, and Non-nuclear Safety (NNS) Class are used instead of Quality Groups A, B, C, and D, respectively, and are consistent with present nuclear industry practice.

2. Paragraph NB-7153 of the ASME Section III Code requires that there be no valves between a code safety valve and its relief point unless special interlocks prevent shutoff without other protection capacity. Therefore, as an alternative to Paragraphs C.1.e and C.2.c, a single safety valve designed, manufactured, and tested in accordance with ASME III Division 1 is considered acceptable as the boundary between the reactor coolant pressure boundary and a lower safety class or NNS class line.

3. Portions of the emergency diesel generator cooling water system, considered by the vendor to be parts of the engine (as distinguished from auxiliary support systems), were built to the manufacturer's standards rather than ASME III. These are identified in Table 3.2-1 and Section 9.5.5. The components used are of high quality, proven by experience, and were designed, fabricated, erected, and tested under the vendor's Quality Assurance Program which meets the requirements of 10CFR50, Appendix B. Similar equipment has been accepted by the NRC for other nuclear power plant applications.

<INSERT Table 1.8.1, end>

RG No. 1.183, Rev. 0
UFSAR Reference Section 15.4.7

ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS, JULY 2000

The assumptions used for evaluating the potential radiological consequences of a fuel handling accident at Beaver Valley Power Station – Unit 2 meet the intent of this regulatory guide.

or 10CFR 50.67, as applicable,

2. 10 CFR 100 dose limit guidelines for potential accidents of exceedingly low probability of occurrence (Chapter 15).

Radioactive waste systems are described in Chapter 11. The radiation monitoring system (RMS) is described in Sections 11.5 and 12.3.4. Waste gas effluents are controlled by charcoal delay beds and, when necessary, waste gases are held in decay tanks until their activity and existing environmental conditions permit their discharge within 10 CFR 20 and 10 CFR 50 requirements. In addition, waste gas effluents are monitored prior to discharge for radioactivity and rate of flow (Section 12.3.4). Section 15.7 describes the results of a rupture of a gas waste storage tank and verifies that the applicable release limits are not exceeded.

Beaver Valley Power Station - Unit 2 liquid waste effluents (Section 11.2) are controlled by batch processing of station radioactive liquids, if necessary, in either BVPS-1 or BVPS-2, sampling before discharge, controlling the rate of release, and prevention of inadvertent tank discharge. Liquid effluents are monitored for radioactivity and rate of flow. Liquid waste disposal system tankage, BVPS-2 steam generator blowdown evaporator capacity, and BVPS-1 liquid waste evaporator capacity are sufficient to handle any expected transient in liquid waste volume.

Station solid wastes (Section 11.4) are prepared in batches in 55-gallon drums and other approved packages for offsite disposal by approved contractors. Solid wastes are shielded for shipment, when necessary, to meet federal regulations.

3.1.2.61 Fuel Storage and Handling and Radioactivity Control (Criterion 61)

3.1.2.61.1 Criterion

"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 (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 filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions."

3.1.2.61.2 Design Conformance

Beaver Valley Power Station - Unit 2 design conforms to the guidelines of GDC 61.

3.1.2.62.2 Design Conformance

The new and spent fuel storage racks are designed in accordance with GDC 62 so that the necessary minimum spacing between nearby fuel assemblies is ensured. The new fuel storage rack accommodates 1/3 of a core plus 17 spare assemblies. The spent fuel storage pool accommodates the spent fuel rack and the required spent fuel shipping cask area.

The fuel racks are arranged so that the spacing between fuel elements cannot be less than that prescribed. Borated water (approximately 2,000 ppm) is used in the spent fuel pool. Even if unborated water were introduced, the spacing is such that criticality is impossible and K_{eff} cannot exceed 0.95. Spent fuel rack design and criticality prevention are fully discussed in Section 9.1.2.

The new fuel assemblies are stored dry in a steel and concrete structure within the fuel building. They are arranged vertically in racks in parallel rows. The steel rack construction prevents possible criticality by requiring that the spacing between fuel elements be not less than that prescribed. The new fuel rack design and criticality prevention is discussed in detail in Section 9.1.1.

Safeguards are provided during handling so that the consequences of the hypothetical worst-case accident ~~(when all fuel rods in a dropped assembly are assumed ruptured)~~ meet 10 CFR ~~100~~ 50.67 guidelines. Chapter 15 provides a complete description of this worst-case accident.

3.1.2.63 Monitoring Fuel and Waste Storage (Criterion 63)

3.1.2.63.1 Criterion

"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."

3.1.2.63.2 Design Conformance

The BVPS-2 design conforms to GDC 63 as follows: Gamma radiation levels in the containment and fuel storage areas are continuously monitored as described in Section 11.5. These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. The fuel pool water temperature is continuously monitored. The temperature is displayed in the main control room where an audible alarm sounds if the water temperature increases above a preset level. Continuous surveillance of radiation levels in the waste storage and handling areas is maintained by ventilation-duct-mounted radiation detectors described in Section 11.5.

TABLE 3.6B-2 (Cont)

Item/Mark No.	Building Location*	Associated Support System**	Required Safety Function***
Steam generator narrow - range level	CS	E,C	Feedwater isolation/ Auxiliary feedwater pump start/ Reactor trip
Steam generator wide - range level	CS	E	Post-accident monitoring
Primary plant demin. water storage tank level	Y	E	Post-accident monitoring/ auxiliary feedwater supply
Containment pressure	CV	E,C	Initiation of safety injection and containment isolation signals on high and high-high containment pressure, post-accident monitoring.
RWST level	Y	E	Post-accident monitoring and transfer to recirculation mode.
Containment sump level	CS	E	Post-accident monitoring and reactor coolant pressure boundary leak detection
Containment sump temperature	CS	E	Post-accident monitoring
HHSI flow/RCS boration	CV,AB	E	Post-accident monitoring
RHS flow	CS	E	Post-accident monitoring
RCS flow	CS	E	Low RCS flow reactor trip
CCP flow	AB	E	Post-accident monitoring
QSS chemical injection	SA	E	Initiate chemical injection for quench spray
Recirculation spray pump minimum flow actuation	SA	E,C	Provide pump protection during accident conditions
Fuel pool level and temperature	FB	E	Post-accident monitoring
LHSI pump recirculation control	SA	E,C	Provide pump protection during accident conditions
Recirculation spray/SWS HX radiaiton monitors	DG	E	Post-accident monitoring
Containment purge radiation monitor	CS	E	Post-accident monitoring ~~prevents~~ controls release of radioactivity from containment
Main steam discharge radiation monitors	MV	E	Post-accident effluent release monitoring

4. The use of redundant demister assemblies, each with an electric heating coil, filter banks, and exhaust fans operable on emergency power.

5. The system is designated as nuclear safety-related.

6 Equipment and system capability of withstanding the design basis earthquake without loss of function.

7. Continuous exhaust through the HEPA filters and charcoal filtration units from the auxiliary building, charging pump cubicles, fuel building, and solid waste handling building. ~~and reactor containment building during refueling.~~

8. Safety-related equipment in the Charging Pump Cubicles, Component Cooling Pump Area, Auxiliary Personnel Airlock, and the Cable Vault Areas Elev. 735' and 755' requires SLCRS flow for cooling.

6.5.3.2.2 System Description

The components and operation of the SLCRS are shown on Figure 6.5-2 and in Tables 6.5-7 and 6.5-8. The primary function of the SLCRS is to ensure that radioactive leakage from the primary containment following a DBA ~~or radioactive release due to a fuel handling accident~~ is collected and filtered for iodine removal prior to discharge to the atmosphere at an elevated release point through a ventilation vent. This ventilation vent also discharges the exhaust from the gland seal steam exhaust system as described in Section 9.4.15.

The SLCRS consists of: 1) one 10,500 cfm and one 29,000 cfm leak collection normal exhaust fans powered from the normal buses, 2) two 34,000 cfm leak collection filter exhaust fans powered from the emergency buses, 3) four 28,500 cfm filter banks, and 4) two 13,000 cfm emergency charging pump cubicle exhaust fans.

Air is exhausted from the fuel building, solid waste handling building, auxiliary building, charging pump cubicles, CCW pump area, post accident sampling system panel and personnel sampling area, and from the areas contiguous to the reactor containment except the main steam and feedwater valve area. The areas contiguous to the reactor containment are the personnel access hatch area, equipment hatch enclosure, purge duct area, main steam and feedwater valve area, cable vault and rod control building at el 735 ft-6 in and el 755 ft-6 in, pipe tunnel, and safeguards areas.

The capacity of each leak collection filter exhaust fan is in excess of the estimated air inleakage to the containment contiguous areas and the other buildings delineated in the previous paragraph. The excess capacity of the fan ensures a negative pressure in the areas being exhausted. Tables 6.5-7 and 6.5-8 and Figures 6.5-2 through 6.5-4 list nominal air flow rates required to ensure the negative pressure.

In order to limit air leakage into these structures to less than the design capacity of a leak collection exhaust fan, all penetration pipes, ducts, and cables are sealed at or near the point where they pass from the contiguous structure to some other structure, such as for example, the auxiliary building. A flexible sealing compound is used between electrical cables and sleeves. Doors are either locked or self-closing, and are under administrative control.

The leak collection filter exhaust fans discharge through a duct to an elevated release point 150 feet above grade. This elevated release point is located on the top of the containment structure, which is 144 feet above grade. The duct and supporting structure is designed to accommodate seismic forces.

The leak collection normal exhaust and filter exhaust fans are also used for reactor containment purging after plant shutdown to remove radioactivity from containment atmosphere. Section 9.4.7, Containment Ventilation System, provides additional information about the purge system.

Should the SLCRS suffer a loss of function, the emergency exhaust fan system, as shown on Figures 6.5-4 and 9.4-4, may be started manually from the main control room to remove the heat generated by the charging pumps and the CCW pumps. The emergency exhaust fan system consists of ducting, motor-operated dampers, and two axial flow fans with back draft dampers located within the tornado missile-protected portion of the auxiliary building. The fans are powered from the emergency buses.

6.5.3.2.3 Safety Evaluation

The SLCRS incorporates redundant 100 percent capacity leak collection exhaust fans, demister assemblies, and main filter banks. In addition, there are redundant dampers where required. The redundant fans, electric heating coils, and dampers are connected to redundant emergency buses, which are capable of being supplied either from normal 4,160 V buses 2A and 2D or emergency diesel generators 2-1 and 2-2 (Figure 8.3-1). Thus, there is sufficient redundancy in the system to ensure system reliability. The SLCRS collects, filters, and releases at an elevated point, the leakage from the containment following a DBA. ~~and leakage from containment with open penetrations and the fuel building following a fuel handling accident.~~ Essentially, all the leakage from the containment following a DBA flows into containment contiguous areas. These areas house the various containment penetrations, ESF equipment circulating radioactive water, and equipment used for plant shutdowns. The SLCRS, with the exception of the ESF portion of the system, is not tornado missile-protected.

The elevated release point in the SLCRS is located above the top of the containment and has a discharge flow rate of about 59,000 cfm. The contiguous area exhaust is normally exhausted directly to atmosphere, but the exhaust is automatically diverted through one of the demister assemblies and main filter banks on an accident signal and is discharged at this elevated release point. Upon failure of both hydrogen recombiners, the hydrogen control system purge blower will take suction from either recombiner suction line. The discharge of the blower is connected directly into the SLCRS contiguous area exhaust ductwork (see Section 6.2.5).

TABLE 6.5-6

PRIMARY CONTAINMENT INFORMATION

Data Description	Parameter Value
Type of structure	Steel-lined reinforced concrete
Primary containment design leak rate	≤ 0.1% per day
Primary containment subatmospheric operation	9.5 psia
Primary containment internal fission product removal systems	
Ice condenser	Not applicable
Spray system (accident)	3,000 gpm each pump
Filter system (normal operation)	2 at 10,000 cfm each
H_2 purge mode (direct; to recirculation systems; to annulus)	
Purge initiation time	Long term backup
Purge rate	50 cfm
Primary containment purge	
Normal plant operation	Containment not purged
~~At refueling~~	~~7,500 cfm max~~
At cold shutdown	29,000 cfm max*
Valve arrangement	Figures 6.5-2 and 6.5-3

*As Built Parameter

TABLE 6.5-8

SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM
AIR FLOW RATES

Modes of Plant Operation	Normal		Normal		Normal		Reactor Trip		Purge		Purge		Refueling		Refueling	
Incident condition	None	None	Rad. Signal in RC Contg. Areas		Loss of Filter Exhaust Fan		DBA or Loss of Normal Power		High Activity in Reactor Cntmnt		No Activity in Reactor Cntmnt		No Activity in Fuel Bldg		High Activity in Fuel Bldg.	
Filtration modes	Filtered	Unfilt	Filtered	Unfilt	Filtered	Unfilt	Filtered	Unfilt	Filtered	Unfilt	Filtered	Unfilt	Filtered	Unfilt	Filtered	Unfilt
Air Flow*(cfm) From:																
Aux bldg - flow path "A" & "B"	39,250	0	26,750	0	17,000	0	0	0	20,500	0	17,000	0	20,500	0	20,500	0
Aux bldg - flow path "C"	14,750	0	14,550	0	15,000	0	16,750	0	12,500	0	15,000	0	12,500	0	12,500	0
Main steam valve area - flow path "D" **	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Fuel bldg - flow path "E"	3,000	0	3,000	0	2,000	0	3,500	0	2,500	0	2,000	0	2,500	0	2,500	0
RC contiguous area - flow path "F"	0	10,500	11,100	0	0	10,500	15,000	0	0	10,500	0	10,500	0	10,500	0	10,500
RC purge - flow path "G"	0	0	0	0	0	0	0	0	9,400***	0	0	29,000	7,500	0	7,500	0
Total flow (cfm):	57,000	10,500	55,400	0	34,000	10,500	35,250	0	44,900	10,500	34,000	39,500	43,000	10,500	43,000	10,500
Exhausted by: exh fan	Filter** exh fan	Normal*** exh fan	Filter exh fan	Normal exh fan	Filter exh fan	Normal exh fan	Filter exh fan	Normal exh fan	Filter exh fan	Normal exh fan	Filter exh fan	Normal exh fan	Filter exh fan	Normal exh fan	Filter exh fan	Normal exh fan
Flow per fan (cfm):																
Fan "A"	28,500	10,500	27,700	0	34,000	10,500	35,250	0	44,900	10,500	34,000	10,500	43,000	10,500	43,000	10,500
Fan "B"	28,500	Standby	27,700	Standby	Lost	Standby	0(single failure	0	0	Standby	0	29,000	Standby		Standby	
Filter Capacity (cfm):																
Main filter - Bank "A"	59,000	-	59,000	0	59,000	-	59,000	-	59,000	-	59,000	-	59,000	-	59,000	-
Main filter - Bank "B"	Standby	-	Standby	-	Standby	-	Standby	-	Standby	-	Standby	-	Standby	-	Standby	-

NOTES:

*For flow paths "A" through "S" refer to Figure 6.5-4.
**Flow path "D" is not used - the duct is blanked off.
***Flow path "G" isolates on High High Activity in Reactor Containment, flow is zero at that time.

Air flow limitation is not required for the purpose of controlling a fuel handling accident radiological release.

TABLE 8.1-2

SAFETY SYSTEMS IDENTIFICATION AND FUNCTION

	Safety Load	Function
1.	The following systems are powered from Class 1E ac sources:	
	High head safety injection system	Emergency core cooling
	Low head safety injection system	Emergency core cooling
	Residual heat removal system	Normal core cooling
	Containment spray systems	Emergency containment cooling
	Primary component cooling water system	Provides cooling water for ESF system and RHRS
	Service water system	Provides cooling of primary component cooling, emergency diesel generators, and safety-related air conditioning
	Auxiliary feedwater system	Provides water to the steam generators when main feedwater is not available
	Spent fuel pool cooling system	Provides cooling for the spent fuel pool
	Safety-related air conditioning and ventilation systems	Provide cooling for Class 1E electrical areas, control areas, and ESF areas
	Post-DBA hydrogen control system	Maintains hydrogen concentration within containment at safe levels following a DBA
	Supplementary leak collection and release system	Collects and filters radioactive leakage from containment following a DBA ~~or a fuel handling accident~~ to prevent release into the atmosphere Collects and releases activity following a fuel handling accident.

The design bases of the spent fuel pool area ventilation system include the following:

1. To maintain the spent fuel pool area indoor air temperature between 76°F and 96°F during normal operation.

2. The spent fuel pool area ventilation system is nonnuclear safety (NNS) class except for the exhaust portion of the ductwork connected to the leak collection system, which is Safety Class 3.

3. General Design Criterion 2, and Regulatory Guide 1.29, as they relate to seismic design classification. The distribution ductwork of the ventilation system is seismically designed and will withstand seismic forces so that the safety-related equipment within the fuel building will not be damaged by ductwork during the postulated seismic events.

4. General Design Criterion 5, as it relates to shared systems and components important to safety. No portion of this system is shared.

5. General Design Criterion 60 and Regulatory Guide 1.52, as they relate to the ability of the system to minimize the release of airborne radioactivity by maintaining the fuel building at negative pressure and by discharging exhaust air through charcoal filtration units which are part of the SLCRS, as described in Section 6.5.3.2.

6. General Design Criterion 61 and Regulatory Guide 1.13, as they relate to spent fuel pool facility design basis controlled leakage during refueling operations. Air flow is maintained from noncontaminated to potentially contaminated areas.

7. During accident conditions, this system is not required to operate, but the SLCRS continues to operate to exhaust and filter air and to maintain a negative pressure in the spent fuel pool area. Filtration following a fuel handling accident is not required to maintain doses within the limits of 10CFR50.67.

9.4.2.2 System Description

The spent fuel pool area ventilation system is shown on Figure 9.4-15. The principal components and design parameters are given in Table 9.4-6.

The recirculation portion of the system includes a 20,000 cfm air handling unit that consists of a roll filter, two chilled water cooling coils, two hot water reheat coils, a fan, controls, and distribution ductwork to all levels.

The exhaust portion of the system, 3,000 cfm, continuously maintains a negative pressure in the building and is connected with the SLCRS.

Condensate from the air handling unit is directed to the fuel pool (Section 9.1.3).

9.4.2.3 Safety Evaluation

normally

The spent fuel pool area ventilation system is nonsafety-related and is not required to operate during accident conditions. The exhaust portion, which is a part of the safety-related SLCRS, continues to operate during a DBA. Air is drawn from the spent fuel pool area, which maintains a negative pressure, and is passed through a filtration unit before being released to the atmosphere. This procedure ensures that the release of airborne radioactivity to the atmosphere is minimized. Airborne radiation levels are measured by a radiation monitor installed in the exhaust duct, which is part of the supplementary leak collection system.

9.4.2.4 Inspection and Testing Requirements

Routine observation and maintenance is performed to ensure operability of the system. The filtration unit housings and filters are components of the supplementary leak collection system and are factory- and site-tested, as described in Section 6.5.1.4. Preliminary tests are performed as described in Section 14.2.12.

9.4.2.5 Instrumentation Requirements

A control switch with indicating lights for the spent fuel pool area air-conditioning unit is provided on the local control panel. Cooling and dehumidification is provided by a chilled water cooling coil thermostatically-controlled valve to maintain a constant discharge air temperature. The reheat coils and the unit space heaters are thermostatically-controlled to maintain the space temperature.

Indication on the local control panel is provided for return air temperature, cooling coils discharge air temperature, supply air temperature, and outdoor air temperature. Annunciators for spent fuel pool area recirculation fan auto trip, spent fuel pool area high temperature, and high radiation in the spent fuel pool area exhaust are provided in the main control room and are also monitored by the BVPS-2 computer system.

9.4.7.3 Containment Purge Air System

The containment purge air system is designed to reduce the airborne radioactivity in the containment after the plant has reached cold shutdown, and to provide outdoor air during extended periods of occupancy such as during refueling, when the containment ~~is~~ may be maintained at a slightly negative pressure.

9.4.7.3.1 Design Bases

The design bases for the containment purge air system are the following:

1. The capacity of the containment purge air system provides approximately one change of containment free air volume every hour. Section 12.3.3 discusses activity levels in the containment following a purge of 8 hours.

2. The containment purge air is supplied at a rate consistent with reducing airborne activity to as low as reasonably achievable.

3. The containment purge supply air is heated or cooled as required.

4. The containment penetrations, the containment isolation valves, and the piping between the valves are Safety Class 2. The remainder of the system is NNS class. The ductwork within the containment building is seismically supported.

5. During refueling, the exhaust air flow ~~is~~ may be reduced as indicated in Table 6.5-8.

9.4.7.3.2 System Description

The containment purge air system is shown on Figure 9.4-9. The performance characteristics are provided in Tables 9.4-7 and 6.5-7. The containment purge air system consists of supply and exhaust air subsystems.

An outdoor air pressure-equalizing line with a manually-operated valve is provided between the isolation valve outside the containment on the purge supply line and its containment penetration. This line is used to bring the containment to atmospheric pressure prior to purging. The purge system is actuated manually after containment is at atmospheric pressure.

The purge air is supplied from one of the auxiliary building air conditioning units, as described in Section 9.4.3.2.

Purge exhaust is provided by the SLCRS, as described in Section 6.5.3.2. The exhaust fans have the capacity to handle approximately one containment air change per hour maintaining the containment under a slightly negative pressure.

The purge exhaust can be manually routed through filter units of the SLCRS ensuring that no radioactivity beyond allowable limits is released to the atmosphere.

A containment airborne monitor provides the plant operators with a high radioactivity alarm. Upon receipt of an alarm the purge exhaust can be manually rerouted through SLCRS filters. Monitors are provided in the purge exhaust ductwork to automatically close the containment purge isolation valves upon detection of high radioactivity in the airstream.

Supply and exhaust ductwork are provided with containment butterfly isolation valves. During normal operation of the plant, the purge supply circuit is inoperative and the isolation valves are closed. Provisions are also made to direct a small quantity of the exhaust air to the BVPS-1 process vent filters and gaseous waste blowers for discharge at the top of the cooling tower, as described in Section 11.3.2.4.

9.4.7.3.3 Safety Evaluation

The isolation valves are closed during subatmospheric containment operation.

Mechanical locking devices are provided to prevent an inadvertent opening of these containment isolation valves.

To prevent discharge of radioactive air to the outside atmosphere, during operation with the containment at atmospheric pressure, air in the containment and in the purge exhaust duct is monitored for radioactivity. High radiation in the containment atmosphere initiates an alarm in the main control room. The operator, upon receipt of the alarm, can reroute the purge exhaust from normal to filtered exhaust and elevated release. Detection of high radioactivity in the purge exhaust airstream will automatically close the containment purge isolation valves, as shown on Figure 6.5 2.

During refueling activities in the containment, the purge may be ~~is~~ routed through the filtered leak collection exhaust at reduced air flow rate as indicated in Table 6.5-8.

This system is not required to operate during accident conditions.

9.4.7.3.4 Inspection and Testing Requirements

The containment isolation valves for the containment purge air system are tested for air-tightness as part of the containment leak testing program (Chapter 16).

t = Time (seconds)

F = Fission rate (fissions per second in fuel region)

α_i = Fission yield for isotope i (atoms per fission)

λ_i = Decay constant for isotope i ($seconds^{-1}$)

γ_i = Escape rate coefficient ($seconds^{-1}$)

$\beta_i = \sigma_{a_i}\phi_{th}$ = Burnup rate ($seconds^{-1}$)

f_{ij} = Branching fraction from i to j

h = Fraction of failed fuel.

The program has a basic library of 167 nuclides with a capability of 200 nuclides. Library data include decay scheme information, production information, and decay gamma spectra in seven energy groups. Input data include time intervals, initial source inventory in the fuel, neutron flux, and power level. The program output describes the system analyzed, as well as the operating history, the activities, and associated gamma spectral information for the input time interval.

The calculation of the core iodine fission product inventory is consistent with the inventories given by DiNunno (et al 1962). The fission product inventories are calculated using the appropriate data from Meek and Rider (1974), Lederer (et al 1968), Nucleonics Handbook of Nuclear Research and Technology (1963), Goldberg (et al 1966), and Perkins (1963). The core iodine and noble gas fission product inventories are presented in Table 11.1-1 based on continuous operation of the unit at 2,766 MWt. These inventories are used in the evaluation of the postulated accidents in Chapter 15.

Fuel assembly source terms for shielding design are calculated using the ACTIVITY 2 computer code and are presented in Chapter 12.

Fuel element heat loadings and stresses, as well as fuel operating experience, are presented in Chapter 4.

11.1.2 Radionuclide Inventory in Fuel Element Gap

The gap activity is that fraction of the gaseous activity in the core that diffuses to the fuel gaps. In accordance with the guidance provided in Regulatory Guide~~s~~ ~~1.25~~ and 1.77, the noble gas and iodine inventory in the fuel gap region is conservatively assumed to be 10 percent ~~(30 percent for Kr-85 for the fuel handling accident analysis)~~ for the accident analysis described in Chapter 15. Table 11.1-1 presents the core gap activities.

For the fuel handling accident, gap activity fractions are from Regulatory Guide 1.183 (0.1 for Kr-85, 0.08 for I-131 and 0.05 for others). Table 15.7-6a presents the associated core gap activities.

4. Inadvertent loading and operation of a fuel assembly in an improper position (Section 15.4.7).

5. Loss-of-reactor-coolant from small ruptured pipes or from cracks in large pipes, which actuate the ECCS (Section 15.6.5).

6. Waste gas system failure (Section 15.7.1).

7. Radioactive liquid waste system leak or failure (atmospheric release) (Section 15.7.2).

8. Liquid containing tank failure (Section 15.7.3).

15.0.1.4 Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to occur, but are postulated because their consequences would include the potential for release of significant amounts of radioactive material. They are the most drastic events which must be designed against and represent limiting design cases. Plant design must be such as to preclude a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault must not cause a consequential loss of required functions of systems needed to mitigate the consequences of the fault including those of the ECCS and containment. The following faults have been classified in this category:

1. Steam system piping failure (Section 15.1.5).

2. Feedwater system pipe break (Section 15.2.8).

3. Reactor coolant pump shaft seizure (locked rotor) (Section 15.3.3).

4. Reactor coolant pump shaft break (Section 15.3.4).

5. Spectrum of RCCA ejection accidents (Section 15.4.8).

6. Steam generator tube failure (Section 15.6.3).

7. Loss-of-coolant accidents (LOCA) resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (RCPB) (Section 15.6.5).

8. Fuel handling accident (Section 15.7.4).

9. Spent fuel cask drop (Section 15.7.5).

or 10 CFR 50.67, as applicable

steam dump, feedwater control, and pressurizer pressure control. The ECCS, including the accumulators, is also modeled.

LOFTRAN is a versatile program which is suited to both accident evaluation and control studies as well as parameter sizing.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits illustrated on Figures 15.0-1 and 15.0-1a. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed by Burnett (1972).

15.0.11.3 TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, or three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2,000 spatial points, and performs its own steady state initialization. Aside from basic cross section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, and control rod motion. Various edits are provided, for example, channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described by Risher and Barry (1975).

15.0.11.4 THINC

The THINC Code is described in Section 4.4.

15.0.12 Radiological Consequences

The radiological consequences of each of the design basis accidents (DBA) were analyzed based on assumptions discussed in the respective sections. Specific parameters used in these analyses are tabulated in the corresponding sections.

Initial core and core gap activities, coolant Technical Specification equilibrium concentrations, pre-accident iodine spike primary coolant concentrations, and concurrent iodine spiking appearance rates are discussed in Section 15.0.9. Coolant concentrations, at design basis and **Technical Specification limit, used in design basis accident radiological consequence analyses prior to 12/98 are given in Section 11.1. Subsequent to 12/98, these values were recalculated using updated plant design and operating parameters and the ORIGENS computer code. The revised values are provided in Table 15.0-8b.** ~~The releases~~

~~to the environment resulting from each accident are presented in the respective sections.~~

Accident atmospheric dispersion coefficients (X/Q) for the exclusion area boundary and low population zone were used to calculate the potential offsite doses. The 0.5 percent sector-dependent X/Q values, presented in Table 15.0-11~~,~~ were determined as described in Section 2.3.4. Main control room X/Q values for the LOCA are also given in Table 15.0-11.
and Table 15.0-14 and Table 15.0-14

The atmospheric releases **discussed** in each accident section are used in conjunction with the appropriate X/Q values of Table 15.0-11 to calculate the potential offsite doses for the corresponding accidents and the potential control room dose due to a LOCA. The methodology for determining the doses is discussed in Appendix 15A. The resulting EAB and LPZ doses are presented in Table 15.0-12 for all postulated accidents. The potential doses to main control room personnel due to DBAs are presented in Table 15.0-13.

or 10 CFR 50.67, as applicable
For all cases the potential offsite doses are within the limits of 10 CFR 100, while the potential doses for the main control room due to a LOCA are within the limits of GDC 19 of Appendix A to 10 CFR 50.
or 10 CFR 50.67, as applicable

15.0.13 References for Section 15.0

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TABLE 15.0-7b

IODINE AND NOBLE GAS INVENTORY
IN REACTOR CORE AND FUEL ROD GAPS*

Nuclide	Core (Ci)	Fraction of Core Activity in Gap	Activity in Fuel Rod Gap (Ci)
I-131	7.21E7	0.12	8.65E6
I-132	1.06E8	0.1	1.06E7
I-133	1.48E8	0.1	1.48E7
I-134	1.63E8	0.1	1.63E7
I-135	1.41E8	0.1	1.41E7
Kr-83m	8.47E6	0.1	8.47E5
Kr-85m	1.73E7	0.1	1.73E6
Kr-85	7.17E5	0.3	2.15E5
Kr-87	3.45E7	0.1	3.45E6
Kr-88	4.78E7	0.1	4.78E6
Kr-89	5.91E7	0.1	5.91E6
Xe-131m	9.80E5	0.1	9.80E4
Xe-133m	4.68E6	0.1	4.68E5
Xe-133	1.48E8	0.1	1.48E7
Xe-135m	3.10E7	0.1	3.10E6
Xe-135	4.06E7	0.1	4.06E6
Xe-137	1.35E8	0.1	1.35E7
Xe-138	1.25E8	0.1	1.25E7

NOTES:

* Based on 1500 days of operation at 2,705 MWt.

Kr-85 and I-131 gap activity fractions are in accordance with ~~Regulatory Guide 1.25 and~~ NUREG/CR-5009.

This table is applicable to design basis accident radiological consequence analyses performed subsequent to December 1998.

Refer to Table 15.7-6a for fuel assembly activity and gap activities use for the fuel handling accident radiological analysis.

TABLE 15.0-12

POTENTIAL DOSES DUE TO POSTULATED ACCIDENTS
(Rem)

Postulated Accident	FSAR Section	Exclusion Area Boundary Thyroid	Exclusion Area Boundary Whole Body Gamma	Exclusion Area Boundary Beta Skin	Low Population Zone* Thyroid	Low Population Zone* Whole Body Gamma	Low Population Zone* Beta Skin
Main steam line break	15.1.5						
Pre-accident Iodine spike		**1.8E+01**	**< 1E-01**	**< 1E+00**	**2.8E+00**	**< 1E-01**	**< 1E+00**
Concurrent Iodine spike		**2.9E+01**	**< 1E-01**	**< 1E+00**	**1.4E+01**	**< 1E-01**	**< 1E+00**
Loss of nonemergency ac power to the station auxiliaries	15.2.6	1.5×10^{-1}	5.2×10^{-4}	4.1×10^{-4}	2.1×10^{-2}	6.5×10^{-5}	6.8×10^{-5}
Locked rotor	15.3.3	**3.7×10^{1}**	**3.6**	**2.2**	**1.6×10^{1}**	**3.6×10^{-1}**	**2.3×10^{-1}**
Rod ejection	15.4.8						
Containment leakage		4.1×10^{1}	1.9×10^{-1}	6.5×10^{-2}	2.0	9.2×10^{-3}	3.2×10^{-3}
Secondary side		2.2×10^{-1}	5.1×10^{-1}	3.7×10^{-1}	1.1×10^{-2}	2.5×10^{-2}	1.8×10^{-2}
Small line break - loss-of-coolant	15.6.2	1.6×10^{1}	7.0×10^{-2}	2.4×10^{-2}	8.2×10^{-1}	3.4×10^{-3}	1.2×10^{-3}
Steam generator tube rupture	15.6.3						
Pre-accident iodine spike		71.6	2.0×10^{-1}	1.0×10^{-1}	3.6	7.0×10^{-3}	5.0×10^{-3}
Concurrent iodine spike		13.4	2.0×10^{-1}	2.0×10^{-1}	8.0×10^{-1}	9.0×10^{-3}	7.0×10^{-3}
Loss-of-coolant	15.6.5						
Containment leakage		2.7×10^{2}	5.3	2.5	1.3×10^{1}	2.6×10^{-1}	1.2×10^{-1}
ECCS leakage		8.3×10^{-1}	1.3×10^{-2}	5.1×10^{-3}	6.3×10^{-1}	1.2×10^{-2}	1.1×10^{-2}
ECCS backleakage to RWST		0.0	0.0	0.0	6.9	7.0×10^{-3}	3.4×10^{-3}
Waste gas system rupture	15.7.1						
Line rupture			3.1×10^{-1}	1.9×10^{-1}			
Tank rupture			1.6×10^{-1}	1.5			
Fuel handling	15.7.4	~~3.8E+01~~	TEDE ~~1.9E+00~~ 2.0E+00		~~1.8E+00~~	TEDE ~~< 1E-1~~ 9.5E-02	

NOTE:

* For duration of accident

TABLE 15.0-13

Control Room Doses, rem, From Design Basis Accidents[7]

Accident	Thyroid	Gamma	Beta	Notes
Main Steam Line Break				
Co-incident Spike	3.0E+00	< 2E-01	< 1E+00	1,4,8
Pre-incident Spike	1.4E+00	< 2E-01	< 1E+00	1,4,8
Small Line Break	8.1	8.0E-4	7.7E-3	3,5
Steam Generator Tube Rupture				
Co-incident Spike	1.9	3.0E-4	6.1E-3	3,5
Pre-incident Spike	8.7	5.0E-4	7.9E-3	3,5
Rod Ejection Accident	4.9	4.9E-4	3.8E-3	3,5
Fuel Handling Accident	~~3.5E+00~~	TEDE ~~< 2E-01~~ 1.4E+00	~~< 1E+00~~	3,9
Locked Rotor Accident	1.7	1.6E-2	2.3E-1	1,4,6
Loss of Auxiliary AC Power	2.1	1.8E-4	1.2E-2	3,5
Waste Gas System Rupture				
Line Break	---	5.8E-2	1.3	3,5
Tank Rupture	---	3.5E-2	9.7	3,5
DBA LOCA	1.3	3.2E-1	1.2E-1	2,5

Notes:

1: Isolation by manual operator action at T=30 minutes post-accident.
2: Control Isolation actuated by CIB signal.
3: No action required.
4: Purge of Control Room atmosphere for 30 minutes at 16,900 cfm at no later than T=8 hr post-accident initiation.
5: Reference: ERS-SFL-93-004
6: Reference: ERS-MPD-91-035
7: Listed dose values represent the bounding value which may be higher than current analysis results.
8: Reference: ERS-SFL-96-010
9: Reference: ERS-SFL-89-019

15.7.4 Radiological Consequence of Fuel Handling Accidents

15.7.4.1 Identification of Causes and Accident Description

The fuel handling accident is classified as an ANS Condition IV event, faults that are not expected to occur but are postulated because their consequences include the potential for the release of significant amounts of radioactive material.

The fuel handling accident is postulated to occur in the fuel building and in the containment. ~~Administrative controls are established to ensure that activity released from either of these areas is filtered (HEPA and charcoal) prior to release to the environment.~~

The fuel handling accident or near sequence of events consists of the maximum dropping of one fuel assembly on ~~other~~ fuel assemblies resulting in cladding damage to the fuel rods ~~in the dropped assembly plus additional rods in the struck assembly~~ with subsequent instantaneous release of all the gap radionuclide inventory. The total number of broken fuel rods is 137, determined by conservative analysis.

The gap radionuclide inventory is based on the minimum time after refueling shutdown of 100 hours and peak inventories for the damaged fuel assemblies. The ~~fuel~~ pool water provides retention capabilities for radiciodines as described in Table 15.7-6.

The radioactivity released from the pool into the building atmosphere ~~is filtered by the supplementary leak collection and release system (Section 6.5.1).~~ will be released via one or a combination of three pathways. These are, the containment equipment hatch, the auxiliary building ventilation system or the SLCRS.

The radioactivity control features of the fuel storage and handling systems inside containment and in the fuel building meet the requirements of GDC 61 (Section 9.1).

15.7.4.2 Analysis of the Effects and Consequences

15.7.4.2.1 Method of Analysis

The assumptions applied to the evaluation of the release of radioactivity from the fuel and the fuel building are based on Regulatory Guide 1.183 ~~1.25, with the exceptions of iodine filter efficiencies which follow the guidance in Regulatory Guide 1.52, and atmospheric dispersion factors, which follow NUREG-0800 (USNRC 1981) (Section 2.3), and I-131 gap activity fraction, which follow NUREG/CR-5009 (USNRC 1988) (Section 3.2.2).~~

~~An evaluation to show that automatic containment isolation occurs upon detection of radioactivity by the redundant containment purge monitors, has been completed. The time required for air to travel from the radiation monitor to the first containment isolation valve is greater than the closure time of the containment isolation valves. The location of the radiation monitors and the containment isolation valves, and the ducting arrangement causes the air travel time to exceed the detector response time plus valve closure time.~~

equipment hatch

Activity may be released to the environment following a fuel handling accident in the containment building through either the open personnel airlock, or other open penetrations. ~~In addition to the administrative controls established to ensure that the activity is collected and filtered prior to release,~~ Controls have been established to close any open penetration following a fuel handling accident. Although this provision will further minimize the activity release, the accident radiological consequence analysis does not take credit for closure.

15.7.4.3 Radiological Consequences

or near

A fuel handling accident is defined as the dropping of one spent fuel assembly onto other assemblies located in either the reactor core or in the fuel building fuel storage racks. ~~The accident is postulated to cause damage to all of the fuel rods in the dropped assembly plus additional rods in other assemblies.~~ The amount of ~~additional~~ damage is dependent upon many factors; however, conservative analyses show that the maximum expected ~~additional~~ damage for the accident occurring either in the fuel building ~~is 50 rods, and in~~ or the reactor containment building ~~353~~ is 137 rods. ~~The atmospheres of either building are filtered (HEPA and charcoal) prior to release to the environment.~~ The analysis is performed assuming that the accident ~~occurs in the containment building,~~ as this represents the bounding case.

release is unfiltered, and is via the auxiliary building ventilation system,

The gap activity in the core fuel assemblies consists of ~~10~~ 5 percent of the core noble gas and iodine activities, except for Kr-85, which is taken as ~~30~~ 10 percent of the Kr-85 core activity and I-131, which is taken as ~~12%~~ 8 of the I-131 core activity at the time of the accident. The damaged fuel assemblies are assumed to have a radial peaking factor of 1.65, ensuring that the analysis addresses the assemblies with the maximum inventory. The gap inventory released into the fuel pool is based on 100 hours of decay resulting from the time between shutdown and movement of the first fuel assembly.

All of the gap activity in the damaged fuel rods is assumed to leak into the pool where 100 percent of the noble gas and ~~1.0~~ 0.5 percent of the iodine is then released into the building. ~~The analysis assumes that activity, which is released into the building atmosphere is exhausted through filters over a 2 hour period. The release to the environment occurs at a point on top of the containment, but for accident evaluation the release is considered to be a ground level release.~~ <INSERT>

The radiological consequences of the postulated fuel handling accident are analyzed based on the assumptions **and initial maximum assembly gap activities** listed in Tables 15.7-6 **and 15.7-6a.** Offsite doses are calculated using the releases in combination with the atmospheric dispersion values given in Table 15.0-14 and the methodology described in Appendix 15A.

The radiological consequences of the postulated fuel handling accident ~~in the fuel building,~~ presented in Table 15.0-12 and Table 15.0-13, are well within the guidelines of 10 CFR ~~100,~~ 50.67 that is, less than ~~75 Rem thyroid and 6 Rem whole body.~~ 25 rem TEDE offsite and 5 rem TEDE for the control room. Additionally, the offsite doses are within the criteria of Regulatory Guide 1.183 and NUREG-0800 of 6.3 rem TEDE.

<INSERT Page 15.7-5>

After becoming airborne, the radioactivity is released to the environment assuming a constant air flow rate (exponential activity removal rate). The analysis model uses a conservatively calculated release rate constant that results in 99.9999% of the activity being released to the environment in the two hours immediately following the accident. Because the accident conditions may include having any of the reactor building containment penetrations open (including the equipment hatch or personnel airlock), and the release may be via any one or a combination of penetrations, the most restrictive release point atmospheric dispersion factor(s) are conservatively applied to the entire release. For the radiological consequence analysis to remain valid, the radioactivity release must be via one of these three points. Additionally, the analysis conservatively does not take credit for any pre-release filtration or iodine plate-out.

15.7.5 Spent Fuel Cask Drop Accidents

15.7.5.1 Identification of Causes and Description

Cask handling procedures ensure that a postulated spent fuel cask drop height of 30 feet is not exceeded. If the spent fuel cask trolley limiting devices are removed during cask handling within the plant, the 30-foot drop height is still not exceeded.

15.7.5.2 Analysis of Effects and Consequences

The details of spent fuel cask handling are provided in Section 9.1.5.

15.7.5.3 Radiological Consequences

Since a spent fuel cask drop exceeding 30 feet cannot occur, no radiological analysis need be performed for a spent fuel cask drop accident.

15.7.6 References for Section 15.7

Underhill, D.W. 1972. Effects of Rupture in a Pressurized Noble Gas Adsorption Bed; Nuclear Safety Volume 13 Number 6.

U.S. Nuclear Regulatory Commission (USNRC 1976). Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). NUREG-0017.

USNRC 1978. Preparation of Radiological Effluent Technical Specification for Nuclear Power Plants. NUREG-0133.

USNRC 1981. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (formerly issued as NUREG 75/087). NUREG-0800.

~~USNRC 1988. Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors. NUREG/CR 5009.~~

BVPS 2000. Unit 2 Fuel Handling Accident Doses at EAB, LPZ, Common Control Room. ERS-SFL-89-019.

USNRC 2000 Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. Regulatory Guide 1.183.

TABLE 15.7-6

ASSUMPTIONS USED FOR THE
FUEL HANDLING ACCIDENT ANALYSIS

Power level (MWt)	2,705
Operating time (days)	1,500
Minimum time since shutdown (hrs)	100
Total number of fuel assemblies in core	157
Number of fuel rods per assembly	264
Fuel damage* (rods)	~~617~~ 137
Fraction of gap activity released	1.0
Radial peaking factor	1.65
Minimum depth of water between top of the damaged fuel rods and fuel pool surface (ft)	23
Fuel pool decontamination factor	
Iodines	~~100~~ 200
Noble gases	1.0
~~Iodine fraction above pool~~	
~~Inorganic~~	~~0.75~~
~~Organic~~	~~0.25~~
~~Fuel building filter efficiency (%)~~	
~~Inorganic~~	~~95~~
~~Organic~~	~~95~~
Offsite breathing rate (m^3/s)	3.5E-04
Release duration (hrs)	2
Control room air intake and exhaust (CFM)	500
Control room volume (ft^3)	1.73E+05
Control room occupancy (fraction)	
0-24 hours	1.0
24 hours - 4 days	0.6
4-30 days	0.4

~~* Because of the differences in the physical configuration, maximum damage for a dropped fuel assembly in containment (617 rods) is greater than that for a dropped assembly in the fuel building (314 rods). Because other accident conditions are similar, the containment accident represents the bounding condition for a fuel handling accident.~~

Control room operator breathing rate (m^3/s)	3.5E-04

TABLE 15.7-6a

ACTIVITIES USED FOR THE FUEL HANDLING ACCIDENT ANALYSIS

	100 hour decay fuel assembly (curies)*	100 hour decay gap activity (curies)**
Kr-85m	2.32E-02	~~3.83E-03~~ 1.91E-03
Kr-85	6.60E+03	~~3.27E+03~~ 1.09E+03
Xe-131m	6.41E+03	~~1.06E+03~~ 5.29E+02
Xe-133m	1.22E+04	~~2.02E+03~~ 1.01E+03
Xe-133	6.56E+05	~~1.08E+05~~ 5.41E+04
Xe-135m	3.85E+00	~~6.35E-01~~ 3.17E-01
Xe-135	1.27E+03	2.09E+02 1.05E+02
I-131	3.~~83~~38E+05	~~6.69E+04~~ 4.46E+04
I-132	2.81E+05	~~4.64E+04~~ 2.32E+04
I-133	3.47E+04	~~5.73E+03~~ 2.86E+03
I-135	2.36E+01	~~3.90E+00~~ 1.95E+00

* Based on core average activity, 1500 days of full power operation at 2705 MWt. The maximum activity for each nuclide was determined considering upper and lower expected bounds of core average enrichment, and average and maximum burn assembly activity.

** Gap activity derivation includes 1.65 multiplier for radial peaking and ~~0.10 activity fraction in gap (Kr-85 gap fraction is 0.30 per Safety Guide 25, I-131 gap fraction is 0.12 per NUREG/CR-5009).~~ activity in gap fractions of 0.1 for kr-85, 0.08 for I-131 and 0.05 for others.

For the control room dose analyses, the EDE is corrected to account for the finite volume of the control room using the method of Murphy-Campe:

$$D_{EDE_{CR}} = \chi / Q \bullet \frac{V^{0.338}}{1173} \bullet \sum_{i} \left(Q_i \bullet C_{EDE_i} \right) \qquad (15A\text{-}8)$$

where:

$D_{EDE_{CR}}$ = Effective Dose Equivalent (EDE) for control room

<INSERT #1>

V = volume of control room, ft^3

Skin Dose Equivalent (skin DE) as described in ICRP-26, replaces the traditional beta skin dose quantity. Based on immersion in a semi-infinite cloud.

$$D_{SKIN} = \chi / Q \bullet \sum_{i} \left(Q_i \bullet C_{SKIN_i} \right) \qquad (15A\text{-}9)$$

where:

D_{SKIN} = Effective Dose Equivalent (skin DE)
Q_i = Activity of nuclide i released
χ/Q = Atmospheric dispersion factor
C_{SKIN_i} = Dose conversion factor for nuclide i (DOE/EH-0070, 1988)

Thyroid Committed Dose Equivalent (thyroid CDE) as described in ICRP-26 and ICRP-30, replaces the traditional thyroid dose quantity based on the critical organ model of ICRP-2 used in TID-14844.

$$D_{CDE_{thy}} = \chi / Q \bullet \sum_{i} \left(Q_i \bullet C_{CDE_i} \bullet BR \right) \qquad (15A\text{-}10)$$

where:

$D_{CDE_{thy}}$ = Thyroid Committed Dose Equivalent (CDE)

Q_i = Activity of iodine isotope i released

χ/Q = Atmospheric dispersion factor
BR = Breathing rate

= 3.47E-4 m^3/sec, 0-8 hours
= 1.75E-4 m^3/sec, 0-24 hours
= 2.32E-4 m^3/sec, > 24 hours
= 3.47E-4 m^3/sec, 0-30 days control room analysis

C_{CDE_i} = Dose conversion factor for nuclide i (USEPA FGR11, 1988)

<INSERT #2>

<INSERT 1 Page 15A-4>

For the EDE calculation where the guidance of Regulatory Guide 1.183 is used, the dose conversion factors are based upon those provided in Federal Guidance Report 12, in lieu of DOE/EH-0700. The methodology described above remains applicable. All of the radionuclides included in the accident source term, including iodines, are considered in the EDE calculation.

<INSERT 2 Page 15A-4>

Committed Effective Dose Equivalent (CEDE) as described in ICRP-26 and ICRP-30. This value represents the dose due to intake of radioactive material. This dose quantity is calculated and used only when analyses are performed pursuant to Regulatory Guide 1.183.

$$D_{CEDE} = \chi/Q \text{ x } \sum_{i} (Q_i \text{ x } C_{CEDE_i} \text{ x BR})$$

where:

D_{CEDE} = Committed Effective Dose Equivalent

Q_i = Activity of nuclide i released

χ/Q = Atmospheric dispersion factor

BR = Breathing rate

3.5E-04 m^3/sec, 0-8 hours (Offsite)

1.8E-04 m^3/sec, 8-24 hours (Offsite)

2.3E-04 m^3/sec, >24 hours (Offsite)

3.5E-04 m^3/sec, 0-30 days (Control Room)

C_{CEDEi} = Dose conversion factor for nuclide i (from USEPA FGR 11, 1988)

Total Effective Dose Equivalent (TEDE) as described in ICRP-26 and ICRP-30. This dose quantity is calculated and used only when analyses are performed pursuant to Regulatory Guide 1.183.

TEDE = EDE + CEDE

ORIGEN

Program ORIGEN calculates fuel depletion, actinide transmutation, fission product buildup and decay and associated radiation source terms. At BVPS, ORIGEN has been used to develop reactor core inventory, and decayed inventories after various cool down times. These values are used in design basis radiological consequence analyses. This code was developed for the NRC at Oak Ridge National Laboratory. This code is documented as part of the SCALE package in NUREG/CR-0200.

15A.4 References for Section 15A

DiNunno, J. J.; Anderson, F.D.; Baker, R.E.; and Waterfield, R.L. 1962. Calculation of Distance Factors for Power and Test Reactor Sites, TID 14844.

U.S. Atomic Energy Commission (USAEC) 1974. Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors. Regulatory Guide 1.4, Revision 2.

Kocher, D. C., External Dose-Rate Conversion Factors for Calculation of Dose to the Public, DOE/EH-0070, 1988

Eckerman, K. F., et al, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, EPA-520/1-88-020, 1988

ICRP, Recommendations of the International Commission of Radiological Protection, ICRP Publication 26, 1977

ICRP, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, 1979

Murphy, K. G. and Campe, K. W., Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19, published in proceedings of 13th AEC Air Cleaning Conference

BVPS Calculation ERS-SFL-96-004, TRAILS_PC: Transport of Radioactive Material in Linear Systems, PC Version

BVPS Calculation ERS-SFL-96-017 ASCOT_PC: Assessment of Containment Transport, PC Version

BVPS Calculation ERS-SFL-96-001, QAD/CGGP_PC, a Point Kernel Photon Shielding Code With Combinatorial Geometry and Geometric Progression Buildup Factors

BVPS Calculation ERS-SFL-88-020, Combinatiorial Geometry Point Kernal Photon and Neutron Shielding Code, QAD-CG, DLC Version 1.0

US NRC NUREG/CR-0200, ORIGEN-S: Scale System Module to Calculate Fuel depletion, Actinide Transmutation, Fission Product Buildup and Decay, and associated Radiation Source Terms

<INSERT>

<INSERT Page 15A-7>

USNRC Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

K. R. Eckerman and J. C. Ryan, External Exposure to Radionuclides in Air, Water and Soil (Federal Guidance Report 12), EPA 402-R-93-081, 1993

ATTACHMENT A-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 219

The following is a list of the affected pages:

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IX
XIII
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

(Proposed Wording)

INDEX

BASES

(Proposed Wording)

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area (RM-207)	1	(1)	≤ 15 mR/hr	$10^{-1} - 10^{4}$ mR/hr	19
b. Containment					
i. Purge & Exhaust Isolation (RMVS 104 A & B)	~~1~~ 2	~~6~~ (2)	≤ 1.6 x 10^{3} cpm	10 - 10^{6} cpm	22
ii. Area (RM-RM-219 A & B)	2	1,2,3 & 4	≤ 1.5 x 10^{4} R/hr	1 - 10^{7} R/hr	35
c. Control Room Isolation (RM-RM-218 A & B)	2	1,2,3,4, ~~5(4), 6(4) (in either unit)~~ and (4)	≤ .47 mR/hr	$10^{-2} - 10^{3}$ mR/hr	41
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection (RM 215B)	1	1,2,3 & 4	N/A	10 - 10^{6} cpm	20
ii. Particulate Activity RCS Leakage Detection (RM 215A)	1	1,2,3 & 4	N/A	10 - 10^{6} cpm	20
b. ~~Fuel Storage Building Gross Activity (RMVS-103 A & B)~~ Deleted	~~1~~	~~(2)~~	~~≤ 4.0 x 10^{4} cpm~~	~~10 - 10^{6} cpm~~	~~21~~

(Proposed Wording)

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

(1) With fuel in the storage pool or building.

(2) During movement of recently ~~With~~ Irradiated fuel ~~in the storage pool.~~

(3) Above background.

(4) During movement of irradiated fuel assemblies ~~or~~ and during movement of ~~heavy loads~~ fuel assemblies over ~~spent fuel~~ irradiated fuel assemblies.

(5) Nominal range for Ch. 7 and Ch. 9. Alarm set on Ch. 7.

(6) Nominal range for Ch. 7 and Ch. 9. Alarm set on Ch. 9.

(7) Other SPING-4 channels not applicable to this specification.

ACTION STATEMENTS

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - This Action is not used. ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the applicable ACTION requirements of Specification 3.9.12 and 3.9.13.~~

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

a) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b) Return the channel to OPERABLE status within 30 days, or, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

ACTION 41 - a) With the number of Unit 1 OPERABLE channels one less than the Minimum Channels OPERABLE requirement:

1. Verify the respective Unit 2 control room radiation monitor train is OPERABLE within 1 hour and at least once per 31 days.

(Proposed Wording)

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 41 (Continued)

assemblies and movement of fuel assemblies over irradiated fuel assemblies

2. With the respective Unit 2 control room radiation monitor train inoperable, suspend all operations involving movement of irradiated fuel within 1 hour and restore the Unit 1 control room radiation monitor to OPERABLE status within 7 days or isolate the control room from the outside atmosphere by closing all series air intake and exhaust isolation dampers, unless the respective Unit 2 control room radiation monitor train is restored to OPERABLE status within 7 days.

b) With no Unit 1 control room radiation monitors OPERABLE:

1. Verify both Unit 2 control room radiation monitors are OPERABLE within 1 hour and at least once per 31 days.

2. With either Unit 2 control room radiation monitor inoperable, suspend all operations involving movement of irradiated fuel within 1 hour and restore the respective Unit 1 control room radiation monitor train to OPERABLE status within 7 days or isolate the control room from the outside atmosphere by closing all series air intake and exhaust isolation dampers, unless the respective Unit 2 control room radiation monitor train is restored to OPERABLE status within 7 days.

3. With no Unit 2 control room radiation monitors OPERABLE, immediately isolate the combined control room by closing all series air intake and exhaust isolation dampers and be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

(Proposed Wording)

TABLE 4.3-3
RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area (RM 207)	S	R	M	*
b. Containment				
i. Purge & Exhaust Isolation (RMVS 104 A & B)	S	R	M	6 ★★
ii. Area (RM-RM-219 A & B)	S	R	M	1,2,3,& 4
c. Control Room Isolation (RM-RM-218 A & B)	S	R	M###	1,2,3,4, ~~5##,~~ and 6## ~~(in either unit)~~ DELETE
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (RM 215B)	S	R#	M	1,2,3 & 4
ii. Particulate Activity RCS Leakage Detection (RM 215A)	S	R#	M	1,2,3 & 4
b. ~~Fuel Storage Building Gross Activity (RMVS-103 A & B)~~	~~S~~	~~R~~	~~M~~	~~**~~ Deleted

* With fuel in the storage pool or building.

** ~~With irradiated fuel in the storage pool.~~ During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

\# Surveillance interval may be extended to the upcoming refueling outage if the interval between refueling outages is greater than 18 months.

\#\# ~~During movement of irradiated fuel~~ During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.

\#\#\# Control Room intake and exhaust isolation dampers and CREBAPS solenoid valves are not actuated.

SURVEILLANCE REQUIREMENTS (Continued)

2. Cycling each weight or spring loaded check valve testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is < 1.2 psid and opens, when the differential pressure in the direction of flow is > 1.2 psid but less than 6.0 psid.

b. Immediately prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of the applicable cycling test, above, and verification of isolation time.

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE* during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. ~~Verifying that on a Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position.~~ Deleted.

d. Cycling each power operated or automatic valve through at least one complete cycle of full travel and measuring the isolation time.

e. Cycling each weight or spring loaded check valve not testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is < 1.2 psid and opens when the differential pressure in the direction of flow is ≥ 1.2 psid but less than 6.0 psid.

f. Cycling each manual valve not locked, sealed or otherwise secured in the closed position through at least one complete cycle of full travel.

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

(Proposed Wording)

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.1 The control room emergency habitability system is OPERABLE ★ when:

a. Two out of three emergency ventilation subsystems, fans, associated filters and dampers are OPERABLE, and

b. Five bottled air pressurization subsystems consisting of two bottles per subsystem are OPERABLE**, and

c. The series normal air intake and exhaust isolation dampers for both units are OPERABLE, and capable of automatic closure on a CIB and Control Room High Radiation isolation signal, or ~~closed~~ OPERABLE by being secured in a closed position with power removed. ★★★

d. The control room air temperature is maintained ≤ 88°F.

APPLICABILITY: ~~a. With either unit* in~~ MODES 1, 2, 3 and 4, ~~or~~ and ~~b.~~ During ~~movement of~~ irradiated fuel assemblies, and ~~movement or movement of loads~~ over irradiated fuel ~~at either unit and a above, or~~ assemblies. During movement of fuel assemblies

DELETE ~~c. Refer to T.S. 3.9.15 when both units are in either MODES 5 or 6.~~ DELETE

ACTION:

a. With less than two emergency ventilation subsystems, fans, and associated filters OPERABLE, restore at least two subsystems to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

a.1 With an emergency ventilation subsystem inlet isolation damper open and not capable of being closed, the requirements of 3.0.3 are applicable.

---

* Emergency power for only one train of dampers ~~of the Unit in MODES 5 or 6 need not be available~~ is required in MODES 5, 6 and with no fuel assemblies in the reactor pressure vessel.

** The air bottles may be isolated for up to 8 hours for performance of instrumentation and control systems testing.

ADD → ★★★ Automatic actuation on a CIB signal is only required in MODES 1 through 4.

(Proposed Wording)

LIMITING CONDITION FOR OPERATION (continued)

b. With one bottled air pressurization subsystem inoperable, restore five bottled air pressurization subsystems to OPERABLE within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b.1 With less than four bottled air pressurization subsystems OPERABLE, the requirements of 3.0.3 are applicable and movement of ~~irradiated fuel~~ shall be suspended.

c. With one open series normal air intake or exhaust isolation damper inoperable and not capable of closing, restore all series dampers to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c.1 With both series normal air intake or exhaust isolation dampers inoperable and not capable of being closed, the requirements of 3.0.3 are applicable and movement of ~~irradiated fuel or movement of loads over irradiated fuel~~ shall be suspended.

d. With the control room air temperature > 88°F but ≤ 105°F, return the temperature to ≤ 88°F in 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d.1 With the control room air temperature > 105°F, be in at least HOT STANDBY within the next 4 hours and in COLD SHUTDOWN within the following 30 hours.

irradiated fuel assemblies and movement of fuel assemblies over irradiated fuel assemblies

(Proposed Wording)

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day and engine-mounted fuel tanks containing a minimum of 900 usable gallons of fuel,

2. A fuel storage system containing a minimum of 17,500 usable gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6, ~~AND~~ and

During movement of irradiated fuel ~~with no fuel~~ assemblies ~~in the reactor vessel~~, and

During movement of ~~loads~~ fuel assemblies over irradiated fuel ~~with no fuel~~ assemblies ~~in the reactor vessel~~.

DELETE DELETE DELETE

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel assemblies, and movement of ~~loads~~ fuel assemblies over irradiated fuel assemblies until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.6.

(Proposed Wording)

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one of the following trains of A.C. Busses shall be OPERABLE and energized in the specified manner:

a. Train "A" A.C. Emergency Busses consisting of:

1. 4160-Volt Emergency Bus #1AE,
2. 480-Volt Emergency Bus #8N,
3. 120-Volt A.C. Vital Bus #I energized from its associated inverter connected to D.C. Bus #1-1, and
4. 120-Volt A.C. Vital Bus #III energized from its associated inverter connected to D.C. Bus #1-3.

b. Train "B" A.C. Emergency Busses consisting of:

1. 4160-Volt Emergency Bus #1DF,
2. 480-Volt Emergency Bus #9P,
3. 120-Volt A.C. Vital Bus #II energized from its associated inverter connected to D.C. Bus #1-2, and
4. 120-Volt A.C. Vital Bus #IV energized from its associated inverter connected to D.C. Bus #1-4.

and

APPLICABILITY: MODES 5 and 6, ~~AND~~

DELETE

During movement of irradiated fuel

DELETE ~~with no fuel~~ assemblies ~~in the reactor vessel~~, and

fuel assemblies

During movement of ~~loads~~ over irradiated fuel ~~with no fuel~~ assemblies ~~in the reactor vessel~~.

DELETE

ACTION:

With the above required train of A.C. Emergency Busses not fully energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel assemblies, and movement of fuel assemblies ~~loads~~ over irradiated fuel assemblies. Initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

(Proposed Wording)

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, one of the following trains of D.C. electrical equipment and busses shall be OPERABLE and energized in the specified manner:

a. Train "A" (orange) consisting of the following:

1. 125-volt D.C. Busses No. 1-1 & 1-3, and
2. 125-volt D.C. Battery Banks 1-1 & 1-3 and Chargers 1-1 & 1-3.

b. Train "B" (purple) consisting of the following:

1. 125-volt D.C. Busses No. 1-2 & 1-4, and
2. 125-volt D.C. Battery Banks 1-2 & 1-4 and Chargers 1-2 & 1-4.

and

APPLICABILITY: MODES 5 and 6, ~~AND~~

DELETE

During movement of irradiated fuel DELETE ~~with no fuel~~ assemblies ~~in the reactor vessel~~, and

fuel assemblies

During movement of ~~loads~~ over irradiated fuel ~~with no fuel~~ assemblies ~~in the reactor vessel~~.

DELETE

ACTION:

With the above required train of D.C. electrical equipment and busses not fully OPERABLE, immediately suspend all operation involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, and movement of ~~loads~~ over irradiated fuel. Initiate corrective action to restore the required train of D.C. electrical equipment and busses to OPERABLE status as soon as possible.

fuel assemblies assemblies

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt battery bank and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

(Proposed Wording)

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least ~~150~~ 100 hours.

APPLICABILITY: During movement of irradiated fuel assemblies in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than ~~150~~ 100 hours, suspend all operations involving movement of irradiated fuel assemblies in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least ~~150~~ 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel.

(Proposed Wording)

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch closed and held in place by a minimum of four bolts,

b. A minimum of one door in each ~~airlock~~ air lock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, manual valve, or approved functional equivalent, or

2. REPLACE WITH INSERT "1" Exhausting at less than or equal to 7500 cfm through OPERABLE Containment Purge and Exhaust Isolation Valves to OPERABLE HEPA filters and charcoal adsorbers of the Supplemental Leak Collection and Release System (SLCRS).

APPLICABILITY: During DELETE ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within the containment, and During movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving DELETE ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel ~~in~~ assemblies within the containment and movement of fuel assemblies over recently irradiated fuel assemblies within the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required condition DELETE ~~within 150 hours prior to the start of and~~ at least once per 7 days ~~during CORE ALTERATIONS or movement of irradiated fuel in the containment~~.

4.9.4.2 The containment purge and exhaust system shall be demonstrated OPERABLE by:

a. Verifying the flow rate ~~through~~ to the SLCRS at least once per 24 hours when the system is in operation,

b. Testing the Containment Purge and Exhaust Isolation Valves per ~~the applicable portions~~ of Specification ~~4.6.3.1.2~~ 4.9.9, and

c. ~~Testing the SLCRS per Specification 4.7.8.1.~~ The required portions of SLCRS shall be demonstrated OPERABLE per Specification 4.7.8.1 with exception to Item 4.7.8.1.c.2.

(Proposed Wording)

INSERT "1"

Capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System with the containment air being exhausted through this system at a flow rate of $\leq$ 7500 cfm to at least one OPERABLE filtered Supplemental Leak Collection and Release System (SLCRS) train.

REFUELING OPERATIONS

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

DELETE recently assemblies

APPLICABILITY: During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within the containment and During movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the purge and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE ~~within 150 hours prior to the start of and~~ at least once per 7 days ~~during CORE ALTERATIONS~~ by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high-high radiation signal from each of the containment radiation monitoring instrumentation channels, and the isolation time of each system isolation valve is within limits. DELETE

(Proposed Wording)

Containment, and
During movement of fuel assemblies over irradiated fuel assemblies within the containment.

REFUELING OPERATIONS

3/4 9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

irradiated DELETE

APPLICABILITY: During movement of fuel assemblies ~~or control rods~~ within the ~~reactor pressure vessel while in MODE 6~~.

ACTION:

irradiated DELETE

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies ~~or control rods~~ within the ~~pressure vessel~~. The provisions of Specification 3.0.3 are not applicable.

Containment

and movement of fuel assemblies over irradiated fuel assemblies within the containment.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth ~~within 2 hours prior to the start of and~~ at least once per 24 hours ~~thereafter during movement of fuel assemblies or control rods~~.

DELETE

(Proposed Wording)

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

within

APPLICABILITY: ~~Whenever~~ irradiated fuel assemblies ~~are in~~ the storage fuel pool

During movement of

, and
During movement of fuel assemblies over irradiated fuel assemblies within the fuel storage pool.

ACTION:

irradiated

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies ~~and crane operations with loads in~~ the fuel storage ~~areas~~ ~~and restore the water level to within its limit within 4 hours~~. The provisions of Specification 3.0.3 are not applicable.

DELETE

DELETE

pool

within the fuel storage pool and movement of fuel assemblies over irradiated fuel assemblies within

SURVEILLANCE REQUIREMENTS

fuel

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days ~~when irradiated fuel assemblies are in the fuel storage pool~~.

DELETE

(Proposed Wording)

FUEL BUILDING VENTILATION SYSTEM - FUEL MOVEMENT

operating with fuel building exhaust flow

portion of the Supplemental Leak Collection and Release System (SLCRS)

LIMITING CONDITION FOR OPERATION

OPERABLE

3.9.12 The fuel building ~~ventilation system~~ shall be ~~operating~~ and discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers ~~during either:~~ .

DELETE

~~a. Fuel movement within the spent fuel storage pool, or~~

~~b. Crane operation with loads over the spent fuel storage pool.~~

During movement of recently

APPLICABILITY: ~~When~~ irradiated fuel ~~which was decayed less than 60~~ assemblies within

DELETE ~~days is in~~ the fuel storage pool, and

During movement of fuel assemblies over recently irradiated fuel assemblies within the fuel storage pool.

ACTION:

With the requirement of the above specification not satisfied, suspend all operations involving movement of ~~fuel~~ fuel within the storage pool ~~or crane operation with loads over~~ the storage pool. The provisions of Specification 3.0.3 are not applicable.

recently irradiated fuel assemblies

and movement of fuel assemblies over recently irradiated fuel assemblies within

SURVEILLANCE REQUIREMENTS

portion of SLCRS

4.9.12 .1 The fuel building ~~ventilation system~~ shall be verified to be operating ~~with all building doors closed~~ ~~within 2 hours prior to the initiation of and~~ at least once per 12 hours ~~during either fuel movement within the fuel storage pool or crane operation with loads over the fuel storage pool~~.

DELETE

← ADD Surveillance Requirements from Specification 4.9.13

with fuel building exhaust flow discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers and that all fuel building doors are closed (1)

ADD

(1) The fuel building doors may be opened for entry and exit.

(Proposed Wording)

~~FUEL BUILDING VENTILATION SYSTEM - FUEL STORAGE~~ 3/4.9.13 (This Specification number is not used.)

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.13 The fuel building ventilation system shall be OPERABLE.~~

~~APPLICABILITY: Whenever irradiated fuel is in the storage pool.~~

~~ACTION:~~

~~With no fuel building ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one fuel building ventilation system is restored to OPERABLE status. The provisions of Specification 3.0.3 are not applicable.~~

DELETE

~~SURVEILLANCE REQUIREMENTS~~

4.9~~.13~~ .12.2 The fuel building ~~ventilation system~~ portion of SLCRS shall be demonstrated OPERABLE:

~~a. At least once per 31 days by initiating flow through the fuel building ventilation system and verifying that the system operates for at least 15 minutes, and~~

DELETE

a. ~~b.~~ At least once per 18 months by:

~~1. Verifying that on a high-high radiation signal, the system automatically directs its exhaust flow through the HEPA filters and charcoal adsorber banks of the Supplemental Leak Collection and Release System (SLCRS).~~ DELETE

1. ~~2.~~ Verifying that the ventilation system maintains the spent fuel storage pool area at a negative pressure of ≥ 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

b. ~~c.~~ Testing the SLCRS per Specification 4.7.8.1 with the exception to item 4.7.8.1.c.2.

MOVE TO SPECIFICATION 4.9.12

(Proposed Wording)

DELETE

DELETE THIS PAGE

~~3/4.9.15 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.15.1 The control room emergency habitability system is OPERABLE when:~~

~~a. Two out of three emergency ventilation subsystems, fans and associated filters and dampers are OPERABLE, and~~

~~b. Five bottled air pressurization subsystems consisting of two bottles per subsystem are OPERABLE, and~~

~~c. The series normal air intake and exhaust isolation dampers for both units are OPERABLE, and capable of automatic closure on a Control Room High Radiation isolation signal*, or closed.~~

~~APPLICABILITY: When both units are in either MODE 5 or 6.~~

~~ACTION:~~

~~a. With less than two emergency ventilation subsystems, fans and associated filters OPERABLE and irradiated fuel being moved or movement of loads over irradiated fuel, restore at least two subsystems to OPERABLE status within 7 days or close at least one series normal air intake and exhaust isolation damper on each intake and exhaust to the control room.~~

~~b. With one bottled air pressurization subsystem inoperable, restore five bottled air pressurization subsystems to OPERABLE within 7 days or suspend all operations involving movement of irradiated fuel or movement of loads over irradiated fuel.~~

~~b.1 With less than four bottled air pressurization subsystems OPERABLE or no emergency ventilation subsystems OPERABLE, suspend all operations involving movement of irradiated fuel or movement of loads over irradiated fuel.~~

~~c. With one open series normal air intake or exhaust isolation damper inoperable# and not capable of closing and irradiated fuel being moved or movement of loads over irradiated fuel, restore all series dampers to OPERABLE status within 7 days or close at least one series normal air intake and exhaust isolation damper on each intake and exhaust to the control room.~~

~~* Not applicable when output relay fuses are removed to prevent inadvertent ESF actuation for a single unit.~~
~~# Emergency backup power not required for any 1 of 2 series dampers.~~

(Proposed Wording)

DELETE

DELETE THIS PAGE

~~ACTION (Continued)~~

~~c.1 With both series normal air intake or exhaust isolation dampers inoperable# and not capable of being closed, suspend all operations involving movement of irradiated fuel or movement of loads over irradiated fuel.~~

SURVEILLANCE REQUIREMENTS

~~4.9.15.1 The emergency ventilation subsystems and the bottled air pressurization system shall be demonstrated OPERABLE in accordance with Specifications 4.7.7.1.1, 4.7.7.1.2 and 4.7.7.2 with the following exception:~~

~~a. Automatic operation upon receipt of a containment phase B isolation signal is not required.~~

~~# Emergency backup power not required for any 1 of 2 series dampers.~~

(Proposed Wording)

BASES

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: 1) the radiation levels are continually measured in the areas served by the individual channels; 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and 3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," October, 1980.

A "recently" irradiated fuel assembly is fuel that has occupied part of a critical reactor core within the previous 100 hours.

3/4.3.3.2 (This Specification number is not used.)

3/4.3.3.3 (This Specification number is not used.)

3/4.3.3.4 (This Specification number is not used.)

, or 5 rem TEDE, as applicable.

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEM

The OPERABILITY of the control room emergency habitability system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The ambient air temperature is controlled to prevent exceeding the allowable equipment qualification temperature for the equipment and instrumentation in the control room. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50 or 10 CFR 50.67, as applicable.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

The OPERABILITY of the SLCRS provides for the filtering of postulated radioactive effluents resulting from ~~a Fuel Handling Accident (FHA) and form~~ leakage of LOSS OF COOLANT ACCIDENT (LOCA) activity from systems outside of the Reactor Containment building, such as Engineered Safeguards Features (ESF) equipment, prior to their release to the environment. This system also collects potential leakage of LOCA activity from the Reactor Containment building penetrations into the contiguous areas ventilated by the SLCRS except for the Main Steam Valve Room and Emergency Air Lock. ~~The operation of this system was assumed in calculating the postulated offsite doses in the analysis for a FHA.~~ System operation was also assumed in that portion of the Design Basis Accident (DBA) LOCA analysis which addressed ESF leakage following the LOCA, however, no credit for SLCRS operation was taken in the DBA LOCA analysis for collection and filtration of Reactor Containment building leakage even though an unquantifiable amount of contiguous area penetration leakage would in fact be collected and filtered. Based on the results of the analyses, the SLCRS must be OPERABLE to ensure that ESF leakage following the postulated DBA LOCA ~~and leakage resulting from a FHA~~ will not exceed 10 CFR 100 limits.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on sealed source contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 and 3/4.7.11 RESIDUAL HEAT REMOVAL SYSTEM (RHR)

Deleted

BASES

3/4.8.1, 3/4.8.2 A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The ACTION requirements specified in ~~MODES 5 and 6~~ LCO's 3.8.1.2, 3.8.2.2, and 3.8.2.4 address the condition where sufficient power is unavailable to recover from postulated events (i.e. fuel handling accident). Implementation of the ACTION requirements shall not preclude completion of actions to establish a safe conservative plant condition. Completion of the requirements will prevent the occurrence of postulated events for which mitigating actions would be required.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods, 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and 3) sufficient power is available for systems (i.e. ~~Supplemental Leak Collection and Release System~~ Control Room ventilation system) necessary to recover from postulated events in these MODES, e.g. a fuel handling accident.

In Modes 1 through 4, the specified quantity of 17,500 usable gallons required in each storage tank (35,000 total gallons) ensures a sufficient volume of fuel oil that, when added to the specified 900 usable gallon volume in the day and engine-mounted tanks, provides the fuel oil necessary to support a minimum of 7 days continuous operation of one diesel generator at full load (UFSAR Sections 8.5.2 and 9.14). The total volume in each of the tanks is greater due to the tank's physical characteristics.

(Proposed Wording)

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2000 ppm) ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The Limitation of K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core when performing those evolutions with the potential to initiate criticality. Suitable detectors used in place of primary source range neutron flux monitors N-31 and N-32 are recognized as alternate detectors. Alternate detectors may be used in place of primary source range neutron flux monitors as long as the required indication is provided. Since installation of the upper internals does not involve movement of fuel or a significant positive reactivity addition to the core, one primary or alternate source range neutron flux monitor with continuous visual indication in the control room provides adequate neutron flux monitoring capability during this evolution.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

50.67

The requirements on containment penetration closure and operability of the containment purge and exhaust system HEPA filters and charcoal adsorbers ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere within 10 CFR ~~100~~ limits. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from ~~a fuel element rupture~~ based upon the lack of containment pressurization potential while in the REFUELING MODE. ~~Operations of the containment purge and exhaust~~

ADD INSERT "2"

the number of fuel rods assumed to be ruptured in the FHA analysis

(Proposed Wording)

INSERT "2"

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the containment to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless 100 hours of decay has occurred. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

An OPERABLE filtered SLCRS train is required to include only those portions of the system that are necessary to ensure that a filtered exhaust path is available from the required plant areas to HEPA and charcoal adsorbers and then to the elevated release point on top of the containment building.

BASES

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

~~system HEPA filters and charcoal adsorbers and the resulting iodine removal capacity are consistent with the assumptions of the accident analysis~~.

All containment penetrations, except for the containment purge and exhaust penetrations, that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Penetration closure may be achieved by an isolation valve, blind flange, manual valve, or functional equivalent. Functional equivalent isolation ensures releases from the containment are prevented for credible accident scenarios. The isolation techniques must be approved by an engineering evaluation and may include use of a material that can provide a temporary, pressure tight seal capable of maintaining the integrity of the penetration to restrict the release of radioactive material from a fuel element rupture.

3/4.9.5 COMMUNICATIONS

The requirements for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies; 2) each crane has sufficient load capacity to lift a control rod or fuel assembly; and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 (This Specification number is not used.)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE,

(Proposed Wording)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION (Continued)

and 2) sufficient coolant circulation is maintained throughout the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

ADD INSERT "3"

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

THE OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The integrity of the containment penetrations of this system may be ~~is~~ required to restrict the release of radioactive material from the containment atmosphere to acceptable levels which are less than those listed in 10 CFR 50.67 ~~100~~. ~~Applicability in MODE 5, although not an NRC safety requirement, will provide additional protection against small releases of radioactive material from the containment during maintenance activities~~.

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99.5% ~~99%~~ of the assumed ~~10%~~ (8% for iodine 131 and 5% for other iodines) iodine gap activity released from ~~the rupture of an irradiated fuel assembly~~ the number of fuel rods assumed to be ruptured in the fuel handling accident analysis. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 ~~and 3/4.9.13~~ FUEL BUILDING VENTILATION SYSTEM

The limitations on the storage pool ventilation system ensure that all radioactive material released ~~from an irradiated fuel assembly~~, as a result of a fuel handling accident (FHA) within the fuel building involving recently irradiated fuel, will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. ~~The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analysis.~~ The spent fuel pool area ventilation system is non-safety related and only recirculates air through the fuel building. The SLCRS portion of the ventilation system is safety-related and maintains a negative pressure in the

ADD INSERT "4"

(Proposed Wording)

INSERT "3"

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the containment to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless 100 hours of decay has occurred. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

INSERT "4"

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the fuel building to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless 100 hours of decay has occurred. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

3/4.9.12 ~~AND 3/4.9.13~~ FUEL BUILDING VENTILATION SYSTEM (Continued)

fuel building. The SLCRS flow is normally exhausted to the atmosphere without filtering, however, the flow is diverted through the main filter banks by manual actuation or on a high radiation signal.

3/4.9.13 (This Specification is not used.)

3/4.9.14 FUEL STORAGE - SPENT FUEL STORAGE POOL

The requirements for fuel storage in the spent fuel pool ensure that: (1) the spent fuel pool will remain subcritical during fuel storage; and (2) a uniform boron concentration is maintained in the water volume in the spent fuel pool to provide negative reactivity for postulated accident conditions under the guidelines of ANSI 16.1-1975. The value of 0.95 or less for K_{eff} which includes all uncertainties at the 95/95 probability/confidence level is the acceptance criteria for fuel storage in the spent fuel pool.

The Action Statement applicable to fuel storage in the spent fuel pool ensures that: (1) the spent fuel pool is protected from distortion in the fuel storage pattern that could result in a critical array during the movement of fuel; and (2) the boron concentration is maintained at ≥ 1050 ppm (this includes a 50 ppm conservative allowance for uncertainties and 600 ppm for margin) during all actions involving movement of fuel in the spent fuel pool.

The Surveillance Requirements applicable to fuel storage in the spent fuel pool ensure that: (1) the fuel assemblies satisfy the analyzed U-235 enrichment limits or an analysis has been performed and it was determined that K_{eff} is ≤ 0.95; and (2) the boron concentration meets the 1050 ppm limit.

The reracked spent fuel pool consists of discrete Regions 1 and 2 with Region 2 further subdivided and identified as Regions 2 and 3. Region 1 is configured to store fuel with a nominal region average enrichment of 5.0 weight percent (w/o) with individual fuel assembly tolerance of + or - 0.05 w/o U-235. The most reactive of the Westinghouse 17 X 17 STD/Vantage 5H and OFA fuel assemblies yielded a maximum K_{eff} of 0.940 including all biases and uncertainties.

Region 2 racks are designed to store fuel with burnup consistent with its initial enrichment. A table of enrichment and corresponding required burnup is provided in the technical specification. A conservative value of the required burnup is given by the following linear equation:

> Minimum burnup for unrestricted storage in Region 2 in MWD/MTU = 12100 * E% - 20500, where E is the initial enrichment in w/o.

3/4.9.14 FUEL STORAGE - SPENT FUEL STORAGE POOL (Continued)

Storage cells in Region 2, which face the pool wall, are in an area of high neutron leakage and are capable of maintaining the K_{eff} below 0.95 with fuel that does not meet the foregoing burnup restriction. A separate calculation to establish the admissibility of storing low burnup fuel in these cells, designated Region 3, has been performed and a table of enrichment and corresponding required burnup is provided in the technical specification. This calculation was performed using the same analytical models and computer codes which were used in the high density rack design. A conservative value of the required burnup is given by the following linear equation:

> Minimum burnup for fuel storage in Region 3 in MWD/MTU = $- 480 * (E\%)^2 + 12,900 * E\% - 27,400$, where E is the initial enrichment in weight percent.

The maximum reactivity in Region 2 is 0.945 and in Region 3 is 0.946 if all cells are loaded with fuel with minimum allowable burnup. This includes all biases and uncertainties and appropriate allowance for uncertainty in depletion calculations.

~~3/4.9.15 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS~~

~~The OPERABILITY of the control room emergency habitability system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The ambient air temperature is controlled to prevent exceeding the allowable equipment qualification temperature for the equipment and instrumentation in the control room. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the whole body radiation exposure to personnel occupying the control room to 5 rem or less, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.~~

(Proposed Wording)

ATTACHMENT A-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 73

The following is a list of the affected pages:

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	XIII
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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

(Proposed Wording)

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BASES

(Proposed Wording)

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BASES

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DESIGN FEATURES

ADMINISTRATIVE CONTROLS

(Proposed Wording)

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area (2RMF-RQ202)	1	(1)	≤75.8 mR/hr	10^{-1} to 10^{4} mR/hr	19
b. Containment Area (2RMR-RQ206 & 207)	2	1,2,3 & 4	≤2.0×10^{4} R/hr	1 to 10^{7} R/hr	35
c. Control Room Area (2RMC-RQ201 & 202)	2	1,2,3 ~~& 4, 5(4) & 6(4)~~ , 4, and (4)	≤0.476 mR/hr	10^{-2} to 10^{3} mR/hr	46,47
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity (Xe-133) RCS Leakage Detection (2RMR-RQ303B)	1	1,2,3 & 4	N/A	10^{-6} to 10^{-1} µCi/cc	20
ii. Particulate Activity (I-131) RCS Leakage Detection (2RMR-RQ303A)	1	1,2,3 & 4	N/A	10^{-10} to 10^{-5} µCi/cc	20
b. ~~Fuel Building Vent~~					
~~i. Gaseous Activity (Xe-133) (2RMF-RQ301B)~~	~~1~~	~~(2)~~	~~≤7.82×10^{-6} µCi/cc~~	~~10^{-6} to 10^{-1} µCi/cc~~	~~21~~

Deleted

(Proposed Wording)

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT (3)	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS (Continued)					
~~ii. Particulate (I-131) (2RMF-RQ301A)~~	~~1~~	~~(2)~~	~~$\leq 6.70 \times 10^{-9}$ µCi/cc~~	~~10^{-10} to 10^{-5} µCi/cc~~	~~21~~
			DELETE		
c. Noble Gas and Effluent Monitors					
i. Supplementary Leak Collection and Release System					
1) Mid Range Noble Gas (Xe-133) (2HVS-RQ109C)	1	1,2,3&4	N.A.	10^{-4} to 10^{2} µCi/cc	35
2) High Range Noble Gas (Xe-133) (2HVS-RQ109D)	1	1,2,3&4	N.A.	10^{-1} to 10^{5} µCi/cc	35
ii. Containment Purge Exhaust (Xe-133) (2HVR-RQ104A & B)	2	(5)	$\leq 1.01 \times 10^{-3}$ µCi/cc	10^{-6} to 10^{-1} µCi/cc	22
iii. Main Steam Discharge (Kr-88) (2MSS-RQ101A,B & C)	1/SG	1,2,3&4	$\leq 3.9 \times 10^{-2}$ µCi/cc	10^{-2} to 10^{3} µCi/cc	35

(Proposed Wording)

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

Not used.

(1) With fuel in the storage pool or building.

(2) ~~With irradiated fuel in the storage pool.~~

(3) Above background.

(4) During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.

(5) During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

ACTION STATEMENTS

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the applicable ACTION requirements of Specifications 3.9.12 and 3.9.13.~~

This Action is not used.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Return the channel to OPERABLE status within 30 days, or, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

(Proposed Wording)

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area (2RMF-RQ202)	S	R	M	*
b. Containment Area (2RMR-RQ206 & 207)	S	R	M	1, 2, 3, 4
c. Control Room Area (2RMC-RQ201 & 202)	S	R	M	1, 2, 3, 4, 5## & 6## and
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (2RMR-RQ303B)	S	R#	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection (2RMR-RQ303A)	S	R#	M	1, 2, 3 & 4
b. ~~Fuel Building Vent~~ Deleted				
~~i. Gaseous Activity (2RMF-RQ301B)~~	~~S~~	~~R~~	~~M~~	~~**~~
~~ii. Particulate Activity (2RMF-RQ301A)~~	~~S~~	~~R~~	~~M~~	~~**~~

*With fuel in the storage pool or building
~~**With irradiated fuel in the storage pool~~
#Surveillance interval may be extended to the upcoming refueling outage if the interval between refueling outages is greater than 18 months.
##During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.

TABLE 4.3-3 (Continued)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. PROCESS MONITORS (Continued)				
c. Noble Gas Effluent Monitors				
i. Supplementary Leak Collection and Release System (2HVS-RQ109C & D)	S	R	M	1, 2, 3 & 4
ii. Containment Purge Exhaust (2HVR-RQ104A & B)	S	R	M	###
iii. Main Steam Discharge (2MSS-RQ101A, B & C)	S	R	M	1, 2, 3 & 4

recently irradiated

During movement of fuel assemblies within containment

the

and during movement of fuel assemblies over recently irrodiated fuel assemblies within the containment.

(Proposed Wording)

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE* at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position. Deleted.

d. Cycling each power operated or automatic valve through at least one complete cycle of full travel and measuring the isolation time pursuant to Specification 4.0.5.

e. Cycling each weight or spring loaded check valve not testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is < 1.2 psid and opens when the differential pressure in the direction of flow is ≥ 1.2 psid but less than 6.0 psid.

f. Cycling each manual valve not locked, sealed or otherwise secured in the closed position through at least one complete cycle of full travel.

* Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

(next page is 3/4 6-31)

(Proposed Wording)

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 The Control Room Emergency Air Cleanup and Pressurization System comprised of the following shall be OPERABLE:

a. A pressurization filtration unit comprised of two trains of fans and filters, and flow path control dampers.**

b. A bottled air pressurization system comprised of 5 subsystems with two bottles in each subsystem.*

c. Two isolation dampers in series in each of four normal air flow paths (two intake and two exhaust) with each damper OPERABLE by automatic actuation*** or OPERABLE by being secured in a closed position with power removed.

APPLICABILITY: ~~All MODES~~ MODES 1, 2, 3 and 4, and During movement of recently irradiated fuel assemblies, and During movement of fuel assemblies over recently irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3 and 4:

With one train of the pressurization filtration unit, or one subsystem of the bottled air pressurization system, or one of two isolation dampers in series inoperable, restore the system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~MODES 5 and 6:~~ During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies:

a. With one train of the pressurization filtration unit or one subsystem of the bottled air pressurization system, or one of two isolation dampers in series inoperable, restore the inoperable system to OPERABLE status within 7 days or suspend all operations involving ~~CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel and movement of loads over irradiated fuel.~~

b. With both trains of the pressurization filtration unit, or more than one subsystem of the bottled air pressurization system, or two of two isolation dampers in series inoperable suspend all operations involving ~~CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel and movement of loads over irradiated fuel.~~

movement of recently irradiated fuel assemblies and movement of fuel assemblies over recently irradiated fuel assemblies.

*The air bottles may be isolated for up to 8 hours for performance of instrumentation and control systems testing.

**Emergency backup power for only one train of dampers and fans of the pressurization filtration unit ~~are~~ is required in MODES 5 ~~and~~ , 6, and with no fuel assemblies in the reactor pressure vessel.

*** Automatic actuation on a CIB signal is only required in MODES 1 through 4.

(Proposed Wording)

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day tank containing a minimum of 350 usable gallons of fuel,

2. A fuel storage system containing a minimum of 53,225 usable gallons of fuel,

3. A fuel transfer pump. and

APPLICABILITY: MODES 5 and 6, ~~AND~~

During movement of recently irradiated fuel DELETE ~~with no fuel~~ assemblies DELETE ~~in the reactor vessel~~, and

During movement of ~~loads~~ fuel assemblies over recently irradiated fuel ~~with no fuel~~ assemblies ~~in the reactor vessel~~. DELETE

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of recently irradiated fuel assemblies, and movement of ~~loads~~ fuel assemblies over recently irradiated fuel assemblies until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.6.

(Proposed Wording)

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one of the following trains of A.C. Busses shall be OPERABLE and energized in the specified manner:

a. Train "A" A.C. Emergency Busses consisting of:

1. 4160-Volt Emergency Bus #2AE,
2. 480-Volt Emergency Bus #2N,
3. 120-Volt A.C. Vital Bus #I energized from its associated inverter connected to D.C. Bus #2-1, and
4. 120-Volt A.C. Vital Bus #III energized from its associated inverter connected to D.C. Bus #2-3.

b. Train "B" A.C. Emergency Busses consisting of:

1. 4160-Volt Emergency Bus #2DF,
2. 480-Volt Emergency Bus #2P,
3. 120-Volt A.C. Vital Bus #II energized from its associated inverter connected to D.C. Bus #2-2, and
4. 120-Volt A.C. Vital Bus #IV energized from its associated inverter connected to D.C. Bus #2-4.

APPLICABILITY: MODES 5 and 6, ~~AND~~ and

DELETE DELETE

During movement of recently irradiated fuel ~~with no fuel~~ assemblies ~~in the reactor vessel~~, and

During movement of ~~loads~~ fuel assemblies over recently irradiated fuel ~~with no fuel~~ assemblies ~~in the reactor vessel~~.

DELETE

ACTION:

With less than the above required train of A.C. Emergency Busses not fully energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of recently irradiated fuel assemblies, and movement of ~~loads~~ fuel assemblies over recently irradiated fuel assemblies. Initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

(Proposed Wording)

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, one of the following trains of D.C. electrical equipment and busses shall be OPERABLE and energized in the specified manner:

a. Train "A" (orange) consisting of the following:

   1. 125-volt D.C. Busses No. 2-1 & 2-3, and
   2. 125-volt D.C. Battery Banks 2-1 & 2-3 and Charger 2-1* & Rectifier 2-3*.

b. Train "B" (purple) consisting of the following:

   1. 125-volt D.C. Busses No. 2-2 & 2-4, and
   2. 125-volt D.C. Battery Banks 2-2 & 2-4 and Charger 2-2* & Rectifier 2-4*.

APPLICABILITY: MODES 5 and 6, ~~AND~~ and

During movement of recently irradiated fuel ~~with no fuel~~ assemblies ~~in the reactor vessel~~, and

During movement of ~~Loads~~ fuel assemblies over recently irradiated fuel ~~with no fuel~~ assemblies ~~in the reactor vessel~~.

DELETE DELETE DELETE

ACTION:

With the above required train of D.C. electrical equipment and busses not fully OPERABLE, immediately suspend all operation involving CORE ALTERATIONS, positive reactivity changes, movement of recently irradiated fuel assemblies and movement of ~~Loads~~ fuel assemblies over recently irradiated fuel assemblies. Initiate corrective action to restore the required train of D.C. electrical equipment and busses to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 125-volt battery bank and chargers/rectifiers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

---

* Spare Charger 2-7 may be substituted for any one charger or rectifier.

(Proposed Wording)

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION OF OPERATION

3.9.3 The reactor shall be subcritical for at least ~~150~~ 100 hours.

APPLICABILITY: During movement of irradiated fuel assemblies in the reactor pressure vessel

ACTION:

With the reactor subcritical for less than ~~150~~ 100 hours, suspend all operations involving movement of irradiated fuel assemblies in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least ~~150~~ 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel.

(Proposed wording)

LIMITING CONDITION FOR OPERATION (Continued)

d. The area(s) outside of containment, where the open containment penetration piping is located, is being exhausted to at least one OPERABLE filtered SLCRS train with all doors to the area(s) required to be serviced by SLCRS closed(1); or

2. Capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System with the containment air being exhausted through this system at a flow rate of ≤ 7500 cfm to at least one OPERABLE filtered SLCRS train.

recently irradiated

APPLICABILITY: During movement of fuel assemblies within the containment and

ACTION: ADD During movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of fuel assemblies ~~in~~ within the containment and movement of fuel assemblies over recently irradiated fuel assemblies within the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required condition:

a. At least once per 7 days, and

b. For all areas located outside of containment containing open containment penetrations, including PAL doors, verify at least once per 12 hours that these areas are being exhausted to filtered SLCRS and that all required area doors are closed.(1)

4.9.4.2 The containment purge and exhaust system shall be demonstrated OPERABLE by:

a. Verifying the flow rate to filtered SLCRS at least once per 24 hours when the system is in operation, and

b. Testing the Containment Purge and Exhaust Isolation Valves per the applicable portions of Specification ~~4.6.3.1.2~~ 4.9.9.

(1) Except for entry and exit.

(Proposed Wording)

REFUELING OPERATIONS

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: During movement of recently irradiated fuel assemblies within the containment, and During movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the purge and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE at least once per 7 days by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation signal from each of the containment radiation monitoring instrumentation channels and the isolation time for each system isolation valve is within limits.

, and During movement of fuel assemblies over irradiated fuel assemblies within the containment.

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

irradiated DELETE

APPLICABILITY: During movement of fuel assemblies ~~or control rods~~ within the containment ~~when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.~~

ACTION:

irradiated DELETE

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies ~~or control rods~~ within the ~~pressure vessel.~~ The provisions of Specification 3.0.3 are not applicable.

containment and movement of fuel assemblies over irradiated fuel assemblies within the containment.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth ~~within 2 hours prior to the start of and~~ at least once per 24 hours ~~thereafter during movement of fuel assemblies or control rods~~.

DELETE

(Proposed Wording)

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: ~~Whenever~~ During movement of irradiated fuel assemblies ~~are in~~ within the fuel storage pool, and During movement of fuel assemblies over irradiated fuel assemblies within the fuel storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of irradiated fuel assemblies ~~and crane operations with loads in~~ within the fuel storage pool and movement of fuel assemblies over irradiated fuel assemblies within the fuel storage pool ~~areas~~ DELETE ~~and restore the water level to within its limit within 4 hours~~. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the fuel storage pool shall be determined to be at least its minimum required depth at least once per 7 days ~~when irradiated fuel assemblies are in the fuel storage pool~~.

DELETE

REFUELING OPERATIONS

FUEL BUILDING VENTILATION SYSTEM - FUEL MOVEMENT

LIMITING CONDITION FOR OPERATION

OPERABLE and — with fuel building exhaust flow

3.9.12 The fuel building portion of the Supplemental Leak Collection and Release System (SLCRS) shall be operating ~~and~~ discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers ~~during either:~~ .

~~a. Fuel movement within the spent fuel pool, or~~ ← DELETE

~~b. Crane operation with loads over the spent fuel storage pool.~~

During movement of recently — assemblies within

APPLICABILITY: ~~When~~ irradiated fuel ~~which was decayed less than 60 days is in~~ the fuel storage pool, and

ACTION: During movement of fuel assemblies over recently irradiated fuel assemblies within the fuel storage pool.

With the requirement of the above specification not satisfied, suspend all operations involving movement of ~~fuel~~ within the storage pool ~~or crane operation~~ DELETE → ~~with loads over the storage pool~~. The provisions of Specification 3.0.3 are not applicable.

recently irradiated fuel assemblies — fuel — and movement of fuel assemblies over recently irradiated fuel assemblies within the fuel storage pool.

SURVEILLANCE REQUIREMENTS

.1

4.9.12 The fuel building portion of the SLCRS shall be verified to be operating ~~with all building doors closed within 2 hours prior to the initiation of and~~ at least once per 12 hours ~~during either fuel movement within the fuel storage pool or crane operation with loads over the fuel storage pool~~. ← DELETE

← ADD Surveillance Requirements from Specification 4.9.13

with fuel building exhaust flow discharging through at least one train of SLCRS HEPA filters and charcoal adsorbers and that all fuel building doors are closed (1)

ADD

(1) The fuel building doors may be opened for entry and exit.

~~FUEL BUILDING VENTILATION SYSTEM - FUEL STORAGE~~

3/4.9.13 (This Specification number is not used.)

DELETE

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.13 The fuel building portion of the Supplemental Leak Collection and Release System (SLCRS) shall be OPERABLE.~~

~~APPLICABILITY: Whenever irradiated fuel is in the storage pool.~~

~~ACTION:~~

~~Without the fuel building portion of the SLCRS OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until the SLCRS portion of the fuel building ventilation system is restored to OPERABLE status. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

4.9.~~13~~ 12.2 The fuel building portion of the SCLRS shall be demonstrated OPERABLE by testing the SLCRS per Specification 4.7.8 with the exception to item 4.7.8.1.c.2.

MOVE TO SPECIFICATION 4.9.12

(Proposed Wording)

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

CHANNEL CALIBRATION

The alternate source range detectors are modified by a note to indicate they are not subject to the source range detector surveillance requirements until they have been connected to the applicable circuits and are required to be OPERABLE. This complies with the testing requirements for components that are required to be OPERABLE.

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. The CHANNEL CALIBRATION for the source range neutron detectors consists of obtaining the detector plateau and preamp discriminator curves, evaluating those curves, and establishing detector operating conditions as directed by the detector manufacturer. The 18 month frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage since performance at power is not possible. The protection and monitoring functions are also calibrated at an 18 month frequency as is normal for reactor protection instrument channels. Operating experience has shown these components usually pass the surveillance when performed on the 18 month frequency.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: 1) the radiation levels are continually measured in the areas served by the individual channels; 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and 3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," October, 1980.

3/4.3.3.2 (This Specification number is not used.)

A "recently" irradiated fuel assembly is fuel that has occupied part of a critical reactor core within the previous 100 hours.

(Proposed Wording)

BASES

3/4.7.5 ULTIMATE HEAT SINK (Continued)

exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants."

3/4.7.6 FLOOD PROTECTION

The limitation on flood level ensures that facility operation will be terminated in the event of flood conditions. The limit of elevation 695 Mean Sea Level was selected on an arbitrary basis as an appropriate flood level at which to terminate further operation and initiate flood protection measures for safety related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM

The OPERABILITY of the control room emergency air cleanup and pressurization system ensures that the control room will remain habitable with respect to potential radiation hazards for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent, or 5 rem TEDE, as applicable. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50 or 10 CFR 50.67, as applicable.

The control room air cleanup system includes two pressurization systems. The filtration pressurization system draws outside air through filters. The bottled air pressurization system pressurizes by discharge of air from bottles without filtration and with closure of intake and exhaust dampers. Although the bottles are shared with Unit 1, the discharge can be initiated by Unit 2 control systems in response to radiation levels. Closure of the intake and exhaust dampers can be initiated by Unit 2 control systems. However, closure of dampers in one intake and in one exhaust is dependent upon availability of Unit 1 power sources.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

The OPERABILITY of the SLCRS provides for the filtering of postulated radioactive effluents resulting ~~from a Fuel Handling Accident (FHA) and~~ from leakage of loss of coolant accident (LOCA) activity from systems outside of the Reactor Containment building, such as Engineered Safeguards Features (ESF) equipment, prior to their release to the environment. This system also collects potential leakage of LOCA activity from the Reactor Containment building

ADD INSERT "1"

(Proposed Wording)

Attachment A-2
Beaver Valley Power Station, Unit No. 2
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INSERT "1"

This LCO is applicable during MODES 1, 2, 3 and 4. This LCO is also applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) for which the requirements of this Specification may be required to limit radiation exposure to personnel occupying the control room. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposure, to personnel occupying the control room, that is within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit personnel exposure. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless 100 hours of decay has occurred. Therefore, this specification will not be applicable, during fuel movement, unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

BASES

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS) (Continued)

penetrations into the contiguous areas ventilated by the SLCRS except for the Emergency Air Lock. ~~The operation of this system was assumed in calculating the postulated offsite doses in the analysis for a FHA.~~ System operation was also assumed in that portion of the Design Basis Accident (DBA) LOCA analysis which addressed ESF leakage following the LOCA, however, no credit for SLCRS operation was taken in the DBA LOCA analysis for collection and filtration of Reactor Containment building leakage even though an unquantifiable amount of contiguous area penetration leakage would in fact be collected and filtered. Based on the results of the analyses, the SLCRS must be OPERABLE to ensure that ESF leakage following the postulated DBA LOCA ~~and leakage resulting from a FHA~~ will not exceed 10 CFR 100 limits.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on sealed source contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 and 3/4.7.11 RESIDUAL HEAT REMOVAL SYSTEM (RHR)

Deleted

(Proposed Wording)

BASES

3/4.8.1, 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

REPLACE WITH INSERT "2"

LCO's 3.8.1.2, 3.8.2.2, and 3.8.2.4

The ACTION requirements specified in ~~MODES 5 and 6~~ address the condition where sufficient power is unavailable to recover from postulated events ~~(i.e., fuel handling accident)~~. Implementation of the ACTION requirements shall not preclude completion of actions to establish a safe conservative plant condition. Completion of the requirements will prevent the occurrence of postulated events for which mitigating actions would be required.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and 3) sufficient power is available for systems (i.e., ~~Supplemental Leak Collection and Release System~~) that may be necessary to recover from postulated events in these MODES, e.g., a fuel handling accident involving recently irradiated fuel.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are based on the recommendations of Regulatory Guides 1.9, Revision 2, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," December 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979,

(Proposed Wording)

INSERT "2"

, such as a fuel handling accident involving recently irradiated fuel. Due to radioactive decay, electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours).

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (2000 ppm) ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

Isolating all reactor water makeup paths from unborated water sources precludes the possibility of an uncontrolled boron dilution of the filled portions of the Reactor Coolant System. This limitation is consistent with the initial conditions assumed in the accident analyses for MODE 6.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core when performing those evolutions with the potential to initiate criticality. Suitable detectors used in place of primary source range neutron flux monitors N-31 and N-32 are recognized as alternate detectors. Alternate detectors may be used in place of primary source range neutron flux monitors as long as the required indication is provided. Since installation of the upper internals does not involve movement of fuel or a significant positive reactivity addition to the core, one primary or alternate source range neutron flux monitor with continuous visual indication in the control room provides adequate neutron flux monitoring capability during this evolution.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure limit leakage of radioactive material within containment to the environment ~~to~~ ensure compliance with 10 CFR ~~100~~ limits. The requirements on operation of the SLCRS ensure that radioactive material released through open containment penetrations, as the result of a fuel

may be required to

ADD INSERT "3"

50.67

(Proposed Wording)

INSERT "3"

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the containment to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless 100 hours of decay has occurred. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

involving recently irradiated fuel,

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

handling accident (FHA) within containment will be filtered through HEPA filters and charcoal absorbers prior to discharge to the atmosphere. These requirements are sufficient to restrict radioactive material release from ~~2.34 fuel assemblies (617 fuel rods) being ruptured as a result of a FHA~~ the number of fuel rods assumed to be ruptured in the FHA analysis based upon the lack of containment pressurization potential while moving fuel assemblies within containment.

Except for the containment purge and exhaust penetrations and open penetrations that meet the requirements of this specification, all containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Penetration closure may be achieved by an isolation valve, blind flange, manual valve, or functional equivalent. Functional equivalent isolation ensures releases from the containment are prevented for credible accident scenarios. The isolation techniques must be approved by an engineering evaluation and may include use of a material that can provide a temporary, pressure tight seal capable of maintaining the integrity of the penetration to restrict the release of radioactive material from a FHA occurring inside containment.

An OPERABLE filtered SLCRS train is required to include only those portions of the system that are necessary to ensure that a filtered exhaust path is available from the required plant areas to HEPA and charcoal adsorbers and then to the elevated release point on top of the containment building. As a minimum, an OPERABLE filtered SLCRS train includes one OPERABLE filtered exhaust fan. If two filtered SLCRS fans are utilized to satisfy the requirements of SR 4.9.4.4, then in order to satisfy the LCO requirements, each fan must be in operation and be OPERABLE with both a normal and emergency power source available.

LCO 3.9.4 requires that a minimum of one train of filtered SLCRS be operating and OPERABLE. A single OPERABLE train of filtered SLCRS that is operating ensures that no undetected failures preventing system operation will occur, and that any active failure will be readily detected. Therefore, the LCO requirement to have an OPERABLE and operating train of filtered SLCRS is sufficient to mitigate the consequences of a FHA within the containment.

The personnel air lock (PAL) area is the plant area where the outer PAL door is located.

(Proposed Wording)

BASES

3/4.9.7 (This Specification is not used.)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained throughout the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

may be

THE OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The integrity of the containment penetrations of this system is required to meet 10 CFR 100 requirements in the event of a fuel handling accident inside containment. The piping that connects this system to filtered SLCRS is not safety related and, therefore, can not be relied upon to mitigate the radiological effects of a fuel handling accident inside containment.

50.67

involving recently irradiated fuel.

ADD INSERT "4"

3/4.9.10 AND 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

99.5%

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from ~~the rupture of an irradiated fuel assembly~~. The minimum water depth is consistent with the assumptions of the accident analysis.

(8% for iodine 131 and 5% for other iodines)

the number of fuel rods assumed to be ruptured in the fuel handling accident analysis

(Proposed Wording)

INSERT "4"

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the containment to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless 100 hours of decay has occurred. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

as a result of a fuel handling accident (FHA) within the fuel building involving recently irradiated fuel,

3/4.9.12 ~~and 3/4.9.13~~ FUEL BUILDING VENTILATION SYSTEM

The limitations on the storage pool ventilation system ensure that all radioactive material released ~~from an irradiated fuel assembly~~ will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. ~~The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analysis.~~ The spent fuel pool area ventilation system is non-safety related and only recirculates air through the fuel building. The fuel building portion of the SLCRS is safety related and continuously filters the fuel building exhaust air. This maintains a negative pressure in the fuel building.

3/4.9.14 FUEL STORAGE - SPENT FUEL STORAGE POOL

The requirements for fuel storage in the spent fuel pool ensure that: (1) the spent fuel pool will remain subcritical during fuel storage; and (2) a uniform boron concentration is maintained in the water volume in the spent fuel pool to provide negative reactivity for postulated accident conditions under the guidelines of ANSI 16.1-1975. The value of 0.95 or less for K_{eff} which includes all uncertainties at the 95/95 probability/confidence level is the acceptance criteria for fuel storage in the spent fuel pool.

Verification that peak fuel rod burnup is less than 60 GWD/MTU is provided in the reload evaluation report associated with each fuel cycle.

The Action Statement applicable to fuel storage in the spent fuel pool ensures that: (1) the spent fuel pool is protected from distortion in the fuel storage pattern that could result in a critical array during the movement of fuel; and (2) the boron concentration is maintained at ≥ 1050 ppm (this includes a 50 ppm conservative allowance for uncertainties) during all actions involving movement of fuel in the spent fuel pool.

The Surveillance Requirements applicable to fuel storage in the spent fuel pool ensure that: (1) the fuel assemblies satisfy the analyzed U-235 enrichment limits or an analysis has been performed and it was determined that K_{eff} is ≤ 0.95; and (2) the boron concentration meets the 1050 ppm limit.

3/4.9.13 (This Specification is not used.)

ADD INSERT "5"

(Proposed Wording)

INSERT "5"

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the fuel building to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless 100 hours of decay has occurred. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 219 and 73
REVISION OF REQUIREMENTS ASSOCIATED WITH CONTAINMENT CLOSURE AND FUEL BUILDING VENTILATION

A. DESCRIPTION OF AMENDMENT REQUEST

For Beaver Valley Power Station (BVPS) Unit No. 1 only:

- The proposed amendment will revise Table 3.3-6 titled "Radiation Monitoring Instrumentation," Item 1.b.i. Specifically, the minimum number of channels required to be operable will be revised from one channel to two channels. In addition, the Applicable Modes specified in Table 3.3-6 for Item 1.b.i will be revised from Mode "6" to table notation (2). The current table notation (2) will be revised to specify the following: "During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment." The Modes in which the surveillance is required as specified in Table 4.3-3 titled "Radiation Monitoring Instrumentation Surveillance Requirements" for Item 1.b.i will be revised from Mode "6" to a footnote designated by **. The current footnote designated by ** will be revised to state the following: "During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment."

- The Applicable Modes specified in Table 3.3-6 for Item 1.c will be revised by deleting the words "(in either unit)".

- Table notation number (4) of Table 3.3-6 will be revised to state the following: "During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies."

- Action 41, specified in Table 3.3-6, will be revised by adding the requirements for suspending all operations involving movement of fuel assemblies over irradiated fuel assemblies. In addition, the term "assemblies" will be added following the word "fuel".

- For Item 1.c contained in Table 4.3-3, the modes in which the applicable surveillances are required will be revised by deleting the words "(in either unit)".

- The current footnote designated by ## contained in Table 4.3-3 will be revised to state the following: "During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies."

- Limiting Condition For Operation (LCO) 3.7.7.1 will be revised by modifying the LCO applicability and required actions. The reference to TS 3.9.15.1 will be deleted. The applicability requirement of "with either unit in" will be deleted. The footnote designated by * will be modified and applied to the term "OPERABLE" in the requirement that the "control room emergency habitability system is OPERABLE". A new footnote designated by *** will be added to LCO 3.7.7.1.c pertaining to control room isolation on a Containment Isolation Phase B (CIB) signal. LCO 3.7.7.1.c will also be revised by replacing the words "or closed" with the words "or OPERABLE by being secured in a closed position with power removed."

- LCO 3.8.1.2 titled " A.C. Sources – Shutdown", LCO 3.8.2.2 titled "A.C. Distribution - Shutdown", and LCO 3.8.2.4 titled "D.C. Distribution – Shutdown", the LCO applicability pertaining to fuel movement will be revised to state the following: "During movement of irradiated fuel assemblies, and During movement of fuel assemblies over irradiated fuel assemblies." The LCO Action requirements will also be revised by adding the word "assemblies" following the word "fuel" and replacing the word "loads" with the words "fuel assemblies".

- LCO 3.9.4 titled "Containment Building Penetrations" will be revised. Specifically, LCO 3.9.4.c.2 will be revised by replacing the existing wording with the words "Capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System with the containment air being exhausted through this system at a flow rate of ≤ 7500 cfm to at least one OPERABLE filtered Supplemental Leak Collection and Release System (SLCRS) train." LCO 3.9.4 will also be revised by removing the terms "Core Alterations or" from the LCO Applicability and Action requirements. In addition, the term "assemblies" will be added to the LCO Applicability and Action requirements following the word "fuel." The word "in" will be replaced with the word "within" following the word "assemblies" in the LCO Action requirement. In addition, the term "recently" will be added to the LCO Mode Applicability and Action requirements preceding the word "irradiated". The LCO Applicability will also be modified by adding the following wording: "and, During movement of fuel assemblies over recently irradiated fuel assemblies within the containment". The action statement will also be modified by adding the words "and movement of fuel assemblies over recently irradiated fuel assemblies within the containment".

- Surveillance Requirement (SR) 4.9.4.1 will be modified by removing the words "within 150 hours prior to the start of and". The words "during Core Alterations or movement of irradiated fuel in the containment" will also be removed from SR 4.9.4.1. SR 4.9.4.2.a will be modified by revising the words "through" to the word "to". SR 4.9.4.2.b will be revised by replacing the reference to Specification 4.6.3.1.2 with a reference to Specification 4.9.9. SR 4.9.4.2.c will be revised to read as follows: "The required portions of SLCRS shall be demonstrated OPERABLE per Specification 4.7.8.1 with the exception to item 4.7.8.1.c.2.

- LCO 3.9.9 titled "Containment Purge and Exhaust Isolation System" will be revised by removal of terms "Core Alterations or" from the LCO Applicability requirements. In addition, the terms "recently" and "assemblies" will be added to the LCO Applicability modifying the words "irradiated fuel." SR 4.9.9 will be revised by removing the words "within 150 hours prior to the start of and" and the words "during CORE ALTERATIONS."

- SR 4.9.13 will be designated as SR 4.9.12.2. SR 4.9.13.a and SR 4.9.13.b.1 will be deleted. The words "ventilation system" in the current SR 4.9.13 will be replaced with the words "portion of SLCRS". SR 4.9.13.b will be designated as SR 4.9.12.2.a. SR 4.9.13.b.2 will be designated as SR 4.9.12.2.a.1. SR 4.9.13.c will be designated as SR 4.9.12.2.b. The current wording of SR 4.9.13.c will be revised by adding the words "with exception to item 4.7.8.1.c.2." following the words "Specification 4.7.8.1". The proposed SR 4.9.12.2 will be moved to Specification 4.9.12.

- Specification 3/4.9.15 titled "Control Room Emergency Habitability Systems" will be deleted.

- Bases Section 3/4.9.9 will also be revised to remove text pertaining to Mode 5 applicability that is not relevant to this specification.

For BVPS Unit No. 2 only:

- Table notation number (4) of Table 3.3-6 will be revised to state the following: "During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.
- Table notation number (5) of Table 3.3-6 will be revised to state the following: "During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment."
- The requirements specified in table notation (2) contained in Table 3.3-6 will be deleted. The words "Not used." will replace this requirement.
- The current footnote designated by ** contained in Table 4.3-3 will be deleted.
- The current footnote designated by ## contained in Table 4.3-3 will be revised to state the following: "During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies."
- The current footnote designated by ### contained in Table 4.3-3 will be revised to state the following: "During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment."
- LCO 3.7.7 will be revised by modifying the LCO applicability and required actions. The footnote designated by ** will be modified. A new footnote designated by *** will be added to LCO 3.7.7.c pertaining to control room isolation on a CIB signal.
- For LCO 3.8.1.2 titled "A.C. Sources – Shutdown", LCO 3.8.2.2 titled "A.C. Distribution - Shutdown", and LCO 3.8.2.4 titled "D.C. Distribution – Shutdown", the LCO applicability pertaining to fuel movement will be revised to state the following: "During movement of recently irradiated fuel assemblies, and During movement of fuel assemblies over recently irradiated fuel assemblies." The LCO Action requirements will also be revised by adding the word "assemblies" following the word "fuel", adding the word "recently" preceding the word "irradiated", and replacing the word "loads" with the words

"fuel assemblies". For LCO 3.8.2.2 only, the word "than" will be added to the LCO Action requirements following the word "less".

- The LCO 3.9.4 Applicability will be modified by adding the following wording: "and, During movement of fuel assemblies over recently irradiated fuel assemblies within the containment". The action statement will also be modified by adding the words "and movement of fuel assemblies over recently irradiated fuel assemblies within the containment". The word "in" will be replaced with the word "within" following the word "assemblies" in the LCO Action requirement. In addition, the term "recently irradiated" will be added to the LCO Mode Applicability and Action requirements preceding the word "fuel". SR 4.9.4.2.b will be revised by replacing the reference to Specification 4.6.3.1.2 with a reference to Specification 4.9.9.

- LCO 3.9.9 titled "Containment Purge and Exhaust Isolation System" will be revised by adding the term "recently irradiated" to the LCO Applicability modifying the word "fuel".

- The current wording of SR 4.9.13 will be revised by adding the words "with exception to item 4.7.8.1.c.2." following the words "Specification 4.7.8". The modified SR 4.9.13 will be designated as SR 4.9.12.2 and moved to Specification 4.9.12.

For BVPS Unit Nos. 1 and 2:

- The Applicable Modes specified in Table 3.3-6 for Item 1.c will be revised from Modes 1,2,3,4,5$^{(4)}$ and 6$^{(4)}$ to Modes 1,2,3,4, and (4).

- For Item 1.c contained in Table 4.3-3, the modes in which the applicable surveillances are required will be revised from Modes 1,2,3,4,5##,6## to Modes 1,2,3,4,and ##.

- The requirements specified in Table 3.3-6 item 2.b pertaining to the fuel building process monitors will be deleted. The word "Deleted" will replace these requirements.

- The requirements specified in Table 4.3-3 item 2.b pertaining to the fuel building process monitors surveillance requirements will be deleted. The word "Deleted" will replace these requirements.

- The requirements specified in Table 3.3-6 Action number 21 will be deleted. The words "This Action is not used." will replace this action requirement.

- SR 4.6.3.1.2.c will be deleted. The word "Deleted." will replace this surveillance requirement.

- LCO 3.9.3 titled "Decay Time" and the associated SR 4.9.3 will be revised by replacing "150 hours" with "100 hours". The word "assemblies" will be added following the word "fuel".

- LCO 3.9.9 Applicability will be modified by adding the following wording: "and, During movement of fuel assemblies over recently irradiated fuel assemblies within the containment".

- SR 4.9.9 will be revised by the addition of a requirement to verify that the isolation time of each system isolation valve is within limits.

- The Applicability for LCO 3.9.10 titled "Water Level – Reactor Vessel" will be revised to read as follows: "During movement of irradiated fuel assemblies within the containment, and During movement of fuel assemblies over irradiated fuel assemblies within the containment." The associated Action requirement will be revised by adding the word "irradiated" preceding the word "fuel", deleting the words "or control rods" and replacing the words "pressure vessel" with the words "containment and during movement of fuel assemblies over irradiated fuel assemblies within the containment."

- SR 4.9.10 will be modified by removing the words "within 2 hours prior to the start of and" and the words " thereafter during movement of fuel assemblies or control rods".

- The Applicability for LCO 3.9.11 titled "Storage Pool Water Level" will be revised to read as follows: "During movement of irradiated fuel assemblies within the fuel storage pool, and During movement of fuel assemblies over irradiated fuel assemblies within the fuel storage pool." The associated Action requirement will be revised by adding the word "irradiated" preceding the word "fuel", replacing the word "areas" with the word "pool", deleting the words "and restore the water level to within its limits within 4 hours" and replacing the words "and crane operations with loads in" with the words "within the fuel storage pool and movement of fuel assemblies over irradiated fuel assemblies within the fuel storage pool."

- SR 4.9.11 will be modified by adding the word "fuel" preceding the word "storage" and removing the words "when irradiated fuel assemblies are in the fuel storage pool".

- LCO 3.9.12 titled "Fuel Building Ventilation System – Fuel Movement" will be revised to state the following: "The fuel building portion of the Supplemental Leak Collection and Release System (SLCRS) shall be OPERABLE and operating with fuel building exhaust flow discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers." The Applicability for LCO 3.9.12 will be revised to read as follows: "During movement of recently irradiated fuel assemblies within the fuel storage pool, and During movement of fuel assemblies over recently irradiated fuel assemblies within the fuel storage pool." The associated Action requirement will be revised by replacing the word "fuel" with the words "recently irradiated fuel assemblies". The words "storage pool" will be replaced with the words "fuel storage pool". The words "or crane operation with loads over the" will be replaced with the words "and movement of fuel assemblies over recently irradiated fuel assemblies within the".

- SR 4.9.12 will be renumbered and become SR 4.9.12.1. In addition, the current surveillance wording will be revised to state the following: "The fuel building portion of SLCRS shall be verified to be operating with fuel building exhaust flow discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers and that all fuel building doors are closed (1) at least once per 12 hours". The words "within 2 hours prior to the initiation of and" and the words "during either fuel movement within the fuel storage pool or crane operation with loads over the fuel storage pool" will be deleted from SR 4.9.12. A new footnote (1) will be added which will modify the requirements of proposed SR 4.9.12.1. This proposed footnote will state the following: "The fuel building doors may be opened for entry and exit."

- LCO 3.9.13 titled "Fuel Building Ventilation System – Fuel Storage" will be deleted. The words "3/4.9.13 (This Specification number is not used.)" will be added in the page header.

- Editorial and format changes are also included. These changes include revision of Index Pages to reflect the deletion of Technical Specifications, changes in

Bases page numbering due to the addition and deletion of text, updating to current page format, the addition of new technical specification pages to accommodate the addition of text, grammatical changes, capitalization changes, word spelling changes, punctuation changes, and the shifting of page footers. The Bases section has been revised as necessary to reflect the changes to these Specifications and the utilization of the Alternative Radiological Source Term and the requirements of 10 CFR 50.67.

B. DESIGN BASES

Fuel handling accident (FHA)

A FHA is classified as an American Nuclear Society Condition IV event, faults that are not expected to occur but are postulated because their consequences include the potential for the release of significant amounts of radioactive material. The FHA is postulated to occur in the fuel building and in containment. A FHA is defined as the dropping of one spent fuel assembly onto another fuel assembly in the spent fuel storage area or in the containment building. Currently, the most limiting FHA is postulated to cause damage to all fuel rods in the dropped assembly (264 rods) plus an additional 34 rods in the struck fuel assembly (BVPS Unit No. 1) and 2.34 assemblies (617 rods) (for BVPS Unit No. 2) with subsequent release of all the activity in the fuel rod gap. The gap inventory released into the pool containing irradiated fuel is based on 150 hours (for BVPS Unit No. 1) and 100 hours (for BVPS Unit No. 2) of decay resulting from the time between shutdown and movement of the first fuel assembly. The bases for the failure of the postulated 298 rods for a FHA in the fuel building is referenced in a Beaver Valley Power Station (BVPS) Unit No. 1 dose calculation as the USNRC staff analysis that indicated that 298 rods (one assembly plus 34 rods) would be damaged. This USNRC staff analysis is also referenced in NUREG 0800, Standard Review Plan (SRP) Section 15.7.4 titled "Radiological Consequences of Fuel Handling Accident." This SRP section states that "The applicant should provide in the SAR conservative analyses for the number of rods assumed damaged both for the spent fuel storage area and inside containment, and the Mechanical Engineering Branch (MEB) should be requested to verify the number of rods assumed damaged. Reference 6 may also be consulted in this regard." Reference 6 is the Long Island Lighting Co., et al., Docket No. STN 50-516/517, further additional supplemental testimony on contention I.D.2 (Spent Fuel Handling Accident) by Walter L. Brooks, et al. The BVPS Unit No. 2 limiting FHA, which assumes the failure of 2.34 assemblies (617 fuel rods), is based on the USNRC staff analysis which states that the maximum number of fuel rods that can be damaged as a result of a dropped fuel assembly in the core is 617 fuel rods or the equivalent of 2.34 fuel assemblies.

Environmental release from the containment through the containment purge and exhaust penetrations is limited by a design which automatically isolates these two containment penetrations following detection of radioactivity by redundant containment purge radiation monitors (RM-1VS-104A & B) for BVPS Unit No. 1 and (2HVS-RQ104A & B) for BVPS Unit No. 2. To verify that a FHA inside containment does not release radioactivity prior to automatic isolation, an evaluation to show that automatic containment isolation occurs upon detection of radioactivity by the redundant containment purge monitors was completed. The time required for air to travel from the radiation monitor to the first containment isolation valve is greater than the closure time of the containment isolation valves. Therefore, the design capability for rapid isolation of the containment provides assurance that minimizes radioactive releases from the containment building via the containment purge and exhaust penetrations. For a FHA in the fuel building, the radioactivity released from the fuel pool into the fuel building atmosphere is filtered by SLCRS.

The assumptions applied to the evaluation of the releases, as a result of a FHA, from the containment and the fuel building are based on Regulatory Guide (RG) 1.25, with the exceptions of iodine filter efficiencies which follow the guidance in RG 1.52; the atmospheric dispersion factors, which follow NUREG-0800 (USNRC 1981) (Section 2.3); and the I-131 gap activity fraction, which follow NUREG/CR-5009 (USNRC 1988) (Section 3.2.2).

BVPS Unit No. 1 SLCRS

The primary function of SLCRS is to ensure that radioactive leakage from the primary containment following a Design Basis Accident (DBA) or radioactive release due to a fuel building FHA is collected and filtered for iodine removal prior to discharge to the atmosphere at an elevated release point through a ventilation vent.

The BVPS Unit No. 1 SLCRS consists of:

1) two exhaust fans powered from the emergency buses (one per train),

2) two filter banks (one per train),

3) two filter bank bypass isolation dampers powered from the emergency buses (one per train)

The supplementary leak collection and release system has two 100 percent design capacity leak collection exhaust fans and two 100 percent design capacity main filter banks. Each main filter bank consists of roughing filters, charcoal filters and HEPA filters. The supplementary leak collection and release system normally exhausts structures contiguous to the containment, the fuel building and the waste gas storage area directly to the elevated release point without the air stream going through the filters. On a containment isolation phase A (CIA) signal, a leak collection area high-high radiation signal, a fuel building high-high radiation signal from the fuel building process radiation monitors (RM-1VS-103A & B), or a waste gas storage area high-high radiation signal, the exhaust from all areas served by the supplementary leak collection and release system is diverted to the filters prior to being discharged at the elevated release point. Should any of these radiation monitoring systems fail, a redundant backup radiation monitor, located in the SLCRS Vent duct, would indicate and alarm to the operator that the preset level had been reached, which would alert the operator to manually divert the exhaust through the filters. The automatic and manual systems are redundant and on separate power supplies.

Air is exhausted from the fuel building, solid waste handling building, charging pump cubicles, containment (during refueling only), certain plant areas contiguous to the reactor containment except for the main steam valve room, and the containment equipment hatch area. The areas contiguous to the reactor containment are the personnel air lock (PAL) area, purge duct area, main steam valve room, cable vaults, piping penetrations area, pipe tunnel, and safeguards areas.

The capacity of each SLCRS fan is in excess of the estimated air inleakage to the containment contiguous areas, fuel building and waste gas storage area. An intake damper is provided at the suction header of the fans to provide makeup air. The excess capacity of the fan ensures a minimum of the required negative pressure in the exhausted areas. Provision is made in the various exhausted areas to permit confirmation that they are maintained at the required negative pressure.

The fuel building ventilation exhaust is monitored by two redundant in-line detectors. A high-high radiation alarm will automatically divert the flow through the prefilter/charcoal/HEPA filter complex of the supplementary leak collection and release system before discharge to the elevated release duct. It will also activate the fuel building evacuation alarm.

The containment purge exhaust is monitored by two redundant inline detectors (RM-1VS-104A & B). During the initial containment purge (prior to allowing personnel entry into containment for refueling operations) an activity alarm will signal the operator to manually actuate damper valves to divert the purge exhaust through the SLCRS filter banks for subsequent discharge via the elevated release point. For the likely no-radiation-alarm condition, the normal purge cycle is completed and the containment is opened to atmosphere permitting entry for refueling operations. During refueling, the containment purge and exhaust system maintains the containment at a slightly negative pressure in accordance with the SLCRS design as stated in UFSAR Section 6.6. Flow will, at all times during refueling, be directed through the SLCRS filters. A high-high activity alarm from either of the containment purge exhaust monitors, will automatically close the purge supply and exhaust isolation dampers in the containment building and activate the containment evacuation alarm. The airborne activity in the containment atmosphere is subsequently discharged at a controlled rate through the SLCRS filters to the elevated release point above the containment building.

If there should be a fuel building high-high radiation signal, the exhaust from the fuel building along with the rest of the leak collection system exhaust, is diverted to one of the main filter banks in the leak collection release system before being released at the SLCRS Vent.

BVPS Unit No. 2 SLCRS

The BVPS Unit No. 2 SLCRS consists of:

1) two unfiltered leak collection normal exhaust fans powered from the normal buses,

2) two filtered exhaust fans powered from the emergency buses,

3) four filter banks (two per train),

4) two emergency exhaust fans powered from the emergency buses to provide unfiltered release and heat removal from the charging pump, component cooling pump area in the event of total loss of SLCRS.

Air is exhausted from the fuel building, solid waste handling building, auxiliary building, charging pump cubicles, component cooling water (CCW) area, post accident sampling panel and personnel sampling area, and from the area contiguous to the reactor containment except the main steam and feedwater valve area. The areas contiguous to the reactor containment are the PAL area, equipment hatch enclosure, purge duct area, main steam and feedwater valve area, cable vault and rod control building at elevation 735 ft-6 in and elevation 755 ft-6 in, pipe tunnel, and safeguards areas.

During normal plant operation, the normal SLCRS exhaust fan is manually set to exhaust unfiltered air from the SLCRS areas, except for the solid waste handling building, auxiliary building, charging pump cubicles, component cooling water (CCW) pump area, and the fuel building. These plant areas are serviced by filtered SLCRS exhaust fans.

On a containment isolation Phase A signal, or on a high radiation signal from monitors in the ventilation exhaust from the areas contiguous to the containment, the air that is normally exhausted by the leak collection normal exhaust fan is diverted so that it first flows through one of the two parallel demister assemblies and then through the aligned main filter banks before flowing to the leak collection filter exhaust fans. The other demister assemblies and main filter banks are used as standby. Each demister assembly consists of a moisture separator and an electric heating coil. Each main filter bank consists of HEPA filter, charcoal, and a second set of HEPA filter. The fuel building is continuously exhausted by filtered SLCRS regardless of the alignment of the normal SLCRS flow paths.

The fuel building airborne radiation monitor draws a sample from the fuel building ventilation exhaust and monitors the radioactivity concentration in the fuel building. It will also activate the fuel building and Decontamination building evacuation alarm.

The containment purge air system is designed to reduce the airborne radioactivity in the containment after the plant has reached cold shutdown, and to provide outdoor air during extended periods of occupancy such as during refueling. The ductwork within the containment building is seismically supported. Redundant radiation monitors (2HVS-RQ104A & B) are located in the containment exhaust ductwork. Upon detecting a predetermined high level of radiation during purging the motor-operated containment isolation valves automatically close.

Purge exhaust is provided by the SLCRS. The exhaust fans have the capacity to handle approximately one containment air change per hour maintaining the containment under a slightly negative pressure.

To prevent discharge of radioactive air to the outside atmosphere, during operation with the containment at atmospheric pressure, air in the containment and in the purge exhaust duct is monitored for radioactivity. High radiation in the containment atmosphere initiates an alarm in the main control room. The operator, upon receipt of the alarm, can reroute the purge exhaust flow from a normal (unfiltered) release path to a filtered release path to the elevated release point on top of the containment building.

BVPS Unit No. 1 CONTROL ROOM VENTILATION

The control room air conditioning system includes two separate systems consisting of two 100 percent redundant air handling units, refrigeration condensing units, river water cooling coils, temperature control air compressors and controls, return air fans, and associated dampers. Each system's electrical components are supplied from separate emergency powered buses.

The main control room is equipped with an emergency pressurization system for use during a DBA. This system is considered essential to prevent radioactive inleakage to the control room by pressurizing the area during the period of time that the containment is above atmospheric pressure (one hour) following a DBA. The requirements of the system are that a sufficient amount of air be released to the control room for a period of 1 hour to maintain a positive pressure in the area thus ensuring a positive air outflow from the isolated area. The system consists of ten Seismic Category I compressed air pressure vessels, each containing 12.5 percent of the required total supply. Thus, there is a 25 percent excess supply equal to the capacity of two storage vessels to satisfy the single failure criteria. There are five Seismic Category I systems of piping and pneumatic controls to control the release of compressed air to the control room area.

Under accident conditions, the control room air-conditioning system outdoor air intake (two dampers in series) and normal exhaust dampers (two dampers in series) are closed at both Units on a CIB signal or by operator action, isolating the entire main control area from the outside atmosphere. At this time, each emergency air pressurization subsystem will have two redundant emergency powered solenoid valves open to release control air from its respective compressed air tank. The operation of either or both of the solenoid valves will cause an isolation valve to open and a pressure control valve to regulate pressure to the metering orifice.

The control room emergency pressurization is used to pressurize the BVPS Unit No. 1 and the BVPS Unit No. 2 main control area for one hour after a design basis accident to protect operating personnel from radioactive inleakage.

Upon depletion of the compressed air supply, the BVPS Unit No. 2 redundant emergency outdoor air pressurization fans and dampers will receive a 60 minutes time delayed CIB signal to operate. The BVPS Unit No. 1 outdoor air pressurization fans and dampers serve as a backup to the BVPS Unit No. 2 outdoor air pressurization fans and dampers and are operated manually as required. Either of the two separate BVPS Unit No. 1 100 percent design capacity emergency outside air pressurization fans may be started as a backup to the BVPS Unit No. 2 fans. The BVPS Unit No. 1 fans draw air from an outside air intake through a single, normally closed, butterfly valve which is provided around the outdoor air butterfly valves and then through redundant, emergency powered heaters, which are provided to limit the relative humidity of the air to the filter banks. The fans then discharge the air through an emergency outside air filter bank consisting of a prefilter, charcoal filter and particulate filter. The filtered air supply is introduced into the air conditioning system to maintain indefinitely the pressurized condition in the main control area. The emergency outside air filters have been provided with a manually operated, normally closed, outlet damper to isolate the control room because the BVPS Unit No. 1 outdoor air pressurization fans and dampers are not controlled automatically.

The BVPS Unit No. 1 and the BVPS Unit No. 2 control room air-conditioning systems are independent and physically separate. The main control room areas are open to each other and are in the same pressure envelope. The control and operation of the two air-conditioning systems are not interconnected. The BVPS Unit No. 1 and the BVPS Unit No. 2 control rooms are permanently occupied and their respective operators may adjust operating parameters with system limitations. The bottled compressed air system is common to both BVPS Unit No. 1 and BVPS Unit No. 2, and its capacity has been sized to accommodate both control rooms.

Power for the emergency bottled air pressurization system can be provided by BVPS Unit No. 1 or BVPS Unit No. 2.

BVPS Unit No. 2 CONTROL ROOM VENTILATION

The main control room air-conditioning system maintains a suitable environment for personnel occupancy and equipment operation during normal and emergency conditions. Two service water cooling coils in the return air stream are also provided as an additional back-up method of cooling the main control room area if required. Components of the air-conditioning systems are seismically designed and are housed in the Seismic Category I control building.

Upon receipt of a CIB signal, or a high radiation signal from the control room area monitors, the normal outside air supply dampers (two dampers in series) and normal exhaust dampers (two dampers in series) automatically close at both Units, thus isolating the control room envelope. This signal also initiates the control room emergency bottled air pressurization system.

The emergency bottled air pressurization system is capable of providing the control room envelope with a sufficient supply of air to meet pressurization requirements. The system is capable of maintaining the ambient pressure slightly above atmospheric pressure, preventing inleakage for approximately 1 hour.

Following the first hour after an accident, the control room envelope is maintained at a positive pressure for an indefinite period of time due to the operation of the redundant emergency supply systems. Each system can draw outside air through a filter assembly which consists of a HEPA filter and carbon adsorber with effective iodine removal efficiency of 95 percent.

The CIB signal isolates the control room almost immediately after a loss of coolant accident (LOCA). For DBAs that do not cause a CIB signal, control room isolation is initiated by a high radiation signal from redundant Category 1 area monitors centrally located in the BVPS-2 control room except for the main steamline break (MSLB) and the locked rotor accident (LRA) which do not initiate a high radiation signal. For the MSLB and the LRA, manual operator action by t=30 min post-accident is needed to maintain habitability. This action consists of purging the control room atmosphere by reconfiguring the control room ventilation system from full recirculation and placing it on full outside air.

In the event of a toxic gas release, main control room habitability is maintained by isolating the air intake, recirculating air conditioned air, and by maintaining a positive pressure using the bottled air system for 1 hour, after which, the main control room will remain isolated for the duration of the accident. The toxic gas protection requirements for the control room are currently specified in the BVPS Licensing Requirements Manual (LRM).

Miscellaneous

A Core Alteration is defined in the BVPS Technical Specifications as the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel.

The PAL is a circular cylinder, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be operable, the door interlock may be disabled allowing both doors of the air lock to remain open. The swinging of the PAL doors is done manually. Each PAL door locking and sealing mechanism is hydraulically driven and actuated by the operation of a pushbutton and associated electric motor driven pump unit. The emergency air lock is located in the containment equipment hatch. This air lock is normally removed from the equipment hatch and stored during a refueling outage. A temporary closure plate is normally installed in place of the emergency air lock when containment closure is required per LCO 3.9.4.

The containment equipment hatch is a circular opening that is stored adjacent to the containment opening via an overhead trolley system. Closing the equipment, to meet the containment closure requirements specified in LCO 3.9.4, involves manually moving the equipment hatch flange in front of the containment opening and lowering the flange into position via a manually operated chain hoist. Once the flange is in the correct position, four swing bolts are moved into position and then manually tightened. No electrical power is required to perform these activities that are necessary to close the equipment hatch. The expedited closure of the containment equipment hatch during shutdown conditions, due to a loss of reactor core decay heat removal, is specified in plant procedures at both Units. The expedited closure of the containment equipment hatch has been demonstrated during a practice drill at BVPS Unit No. 2.

The BVPS waste gas decay tanks are located in underground storage vaults. Each BVPS Unit has separate storage vaults. Each of the decay tanks is considered independently regarding tank bursts for the requirements of 10 CFR 100, in that each tank is shielded from any other tank. Double valving and missile protection are provided on all interconnecting piping.

C. JUSTIFICATION

REVISION OF MINIMUM NUMBER OF OPERABLE CHANNELS (For BVPS Unit No. 1 only)

The proposed change of the minimum number of radiation monitors specified in Table 3.3-6 for Item 1.b.i from "1" to "2" and the change in the applicable Modes aligns the radiation monitor operability requirements with the containment purge isolation requirements specified in LCO 3.9.9, SR 4.9.9, and LCO 3.9.4. This proposed change is consistent with the intent of Specification 3.3.6 titled "Containment Purge and Exhaust Isolation Instrumentation" contained in the NUREG 1431 Revision 1 titled "Standard Technical Specifications - Westinghouse Plants" (ISTS). When the plant is in Modes 5 or 6 without fuel handling in progress in the containment, these radiation monitors do not need to be operable since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within regulatory limits. The operability of these radiation monitors is intended to provide protection against a FHA within containment. The proposed mode applicability will continue to require that two radiation monitors be operable during plant evolutions when a FHA can occur (i.e., during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies) and these radiation monitors are necessary to initiate containment purge and exhaust isolation. The proposed change in the mode in which the surveillance is required, as specified in Table 4.3-3 for Item 1.b.i is justified based on the above discussion for changing the mode applicability specified in Table 3.3-6 for these same radiation monitors. Surveillance Requirements do not need to be performed on equipment that is not required to be operable.

DELETION OF CERTAIN CORE ALTERATIONS REQUIREMENTS

The proposed deletion of the term "Core Alterations" from LCO 3.9.4 and LCO 3.9.9 (for BVPS Unit No.1 only) is justified since a FHA is the only event during Core Alterations that is postulated to result in fuel damage and radiological release. The accidents that are postulated to occur during Core Alterations, in addition to a FHA, are: inadvertent criticality (due to a control rod removal error or continuous control rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. The proposed LCO applicability will continue to require that plant equipment be operable when required to mitigate a FHA during plant evolutions where a FHA can occur inside containment when required; i.e., during movement of recently irradiated fuel assemblies within containment and during movement of fuel assemblies over recently irradiated fuel assemblies within containment. This proposed change to remove Core Alterations from these LCOs is also consistent with NRC approved Technical Specification Traveler Form (TSTF) 51 Revision 2 to the ISTS. This change is also consistent with the changes approved by Amendment number 116 (TAC No. MA8861) to the BVPS Unit 2 Technical Specifications.

Based on the above, the proposed deletion of the terms "or control rods" from Specification 3/4.9.10 is also justified. Movement of control rods within the reactor pressure vessel is a core alteration. These events are not postulated to result in fuel cladding integrity damage. Therefore, maintaining 23 feet of water of the seated fuel is not required during movement of control rods within the reactor pressure vessel. This change is also consistent with NRC approved Technical Specification Traveler Form (TSTF) 51 Revision 2 to the ISTS.

REVISION OF LCO APPLICABILITY AND ACTIONS

The proposed revisions of the BVPS Unit No. 1 LCO applicability and action requirements, specified in Table 3.3-6 for item 1.c, LCO 3.7.7.1, and LCOs 3.8.1.2, 3.8.2.2, and 3.8.2.4 for the control room radiation monitors, for the control room habitability systems, and for the electrical power requirements, respectively, will ensure that the requirements of these specifications apply and appropriate actions are taken during fuel handling activities that involve the potential need for these systems to mitigate the consequences of a FHA in order to meet the radiation exposure limits specified in 10 CFR 50.67. The addition of "and during movement of fuel assemblies over irradiated fuel assemblies" is necessary based on the results of the fuel drop analysis contained in Attachment D of this LAR. The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod failure. The movement of a non-irradiated fuel assembly could result in a radiological release that may require the use of the control room habitability systems if this assembly were to strike an irradiated fuel assembly. Therefore, the adoption of the ISTS LCO applicability wording of "During movement of irradiated fuel assemblies" is not sufficient to meet the BVPS fuel drop analysis.

The proposed revisions of the BVPS Unit No. 2 LCO applicability and action requirements specified in Table 3.3-6 item 1.c, LCO 3.7.7, and LCOs 3.8.1.2, 3.8.2.2, and 3.8.2.4, for the control room radiation monitors, the control room habitability systems, and the electrical power requirements, respectively, will ensure that the requirements of these specifications apply and the appropriate actions are taken during fuel handling activities that involve the potential need for these systems to mitigate the consequences of a FHA in order to meet the radiation exposure limits specified in 10 CFR 50.67. The addition of "and during movement of fuel assemblies over recently irradiated fuel assemblies" is necessary based on the results of the fuel drop analysis contained in Attachment D of this LAR. The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod failure. The movement of a non-irradiated fuel or non-recently irradiated fuel assembly could result in a radiological release that may require the use of the control room habitability systems if this assembly were to strike a recently irradiated fuel assembly. Therefore, the adoption of the ISTS LCO applicability wording of "During movement of irradiated fuel assemblies" is not sufficient to meet the BVPS fuel drop analysis.

The proposed revisions of Table 3.3-6 item (1.b.i for BVPS Unit No.1) / 2.c.ii for BVPS Unit No. 2), LCO 3.9.4, and LCO 3.9.9 applicabilities and actions, to specify during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies, will ensure that requirements of these specifications apply and that appropriate actions are taken during fuel handling activities that involve the potential need for these systems to mitigate the consequences of a FHA in order to meet the radiation exposure limits specified in 10 CFR 50.67. The addition of "and during movement of fuel assemblies over recently irradiated fuel assemblies" is necessary based on the results of the fuel drop analysis contained in Attachment D of this LAR. The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod failure. The movement of a non-irradiated or a non-recently irradiated fuel assembly could result in a radiological release that may require containment isolation and/or filtration if this assembly were to strike a recently irradiated fuel assembly. Therefore, the adoption of the ISTS LCO applicability wording of "During movement of recently irradiated fuel assemblies" is not sufficient to meet the BVPS fuel drop analysis. These LCOs will not be applicable when moving non-irradiated or non-recently irradiated fuel assemblies provided that these fuel assemblies are not moved over recently irradiated fuel assemblies. During these evolutions, neither the dropped fuel assembly nor the stuck assembly has sufficient amount of radioactivity to require filtration prior to a release to the environment. The current BVPS Unit No. 2 LCO applicability of "During movement of fuel assemblies within containment" will be revised to the LCO applicability stated above based on the above discussion on the level of radioactivity contained in a non-irradiated fuel assembly. Currently, the term "Core Alterations" in the LCO applicability (for BVPS Unit No. 1 only) requires that LCO 3.9.4 and LCO 3.9.9 be applicable during the movement of any fuel within the reactor vessel. For BVPS Unit No. 2 only, LCO 3.9.4 and LCO 3.9.9 applicabilities and LCO 3.9.4 actions of movement of fuel within containment are applicable during the movement of any fuel within the reactor vessel. Therefore, LCO 3.9.4 and 3.9.9 are currently applicable during movement of non-irradiated fuel within the reactor vessel.

The proposed revisions of the LCO applicability and action requirements specified in LCO 3.9.10 and LCO 3.9.11 will ensure that the requirements of these specifications apply and that the appropriate actions are taken during fuel handling activities that involve the potential need for these minimum water levels to mitigate the consequences of a FHA in order to meet the radiation exposure limits specified in 10 CFR 50.67. The addition of "and during movement of fuel assemblies over irradiated fuel assemblies" is necessary based on the results of the fuel drop analysis contained in Attachment D of this LAR. The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod failure. The movement of a non-irradiated fuel could result in a radiological release that may require the minimum water level to mitigate the FHA if this assembly were to strike a recently irradiated fuel assembly. Therefore, the adoption of the ISTS LCO applicability wording of "During movement of irradiated fuel assemblies" is not sufficient to meet the BVPS fuel drop analysis.

The proposed revisions of the LCO applicability and action requirements specified in LCO 3.9.12 for the fuel building ventilation system will ensure that requirements of these specifications apply during fuel handling activities that involve the potential need for this system to mitigate the consequences of a FHA in order to meet the radiation exposure limits specified in 10 CFR 50.67. The addition of "and during movement of fuel assemblies over recently irradiated fuel assemblies" is necessary based on the results of the fuel drop analysis contained in

Attachment D of this LAR. The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod failure. The movement of a non-irradiated fuel or non-recently irradiated fuel assembly could result in a radiological release that may require the use of the fuel building ventilation system if this assembly were to strike a recently irradiated fuel assembly. Therefore, the adoption of the ISTS LCO applicability wording of "During movement of irradiated fuel assemblies" is not sufficient to meet the BVPS fuel drop analysis.

USE OF THE TERM "RECENTLY" IN LCO APPLICABILITY AND ACTION REQUIREMENTS

The revised radiological analyses contained in Attachment C of this LAR demonstrates that the consequences of a fuel handling accident once the fuel had undergone radioactive decay for several days will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without reliance on the requirements of these Specifications to limit leakage to the environment and to limit control room personnel exposure. The length of this "several day" period has been determined to be 100 hours by the FHA radiological analysis. The term "recently irradiated" fuel will be defined in the applicable TS Bases sections as "fuel that has occupied part of a critical reactor core within the previous 100 hours". This LAR permits fuel handling activities to occur after this period of radioactive decay, without requiring Technical Specification controls over the fuel building and the containment building integrities and the associated building ventilation system/filtration operability. [Note: This LAR continues to require Technical Specification controls over building integrity and ventilation system operability during the first 100 hours of the outage (when the fuel is still classified as "recently irradiated").]

The addition of the term "recently" to certain LCO applicabilities is based on the radiological analysis contained in Attachment C of this LAR and the guidance provided in TSTF 51 Revision 2. The purpose of the addition of the term "recently" is to establish a point where operability of those systems typically used to mitigate the consequences of a FHA is no longer required to meet the radiation exposure limits specified in 10 CFR 50.67. The LCOs for which the term "recently" will be added applies to those systems (with the exceptions discussed below) which are no longer required to be operable to mitigate a FHA once the irradiated fuel has undergone 100 hours of radioactive decay.

The proposed wording of LCO 3.9.3 will prohibit irradiated fuel movement unless 100 hours of decay has occurred. Therefore, those specifications that will have the term "recently" irradiated (excluding the electrical power Specifications) will not be applicable unless the decay time in Specification 3.9.3 is reduced to below 100 hours based on a revised radiological analysis.

For BVPS Unit No. 1 only, the control room habitability system (i.e., the pressurization bottles, emergency supply fans and damper isolation) is not required to mitigate the consequences of a FHA (based on the proposed radiological analysis) in order to maintain the radiation doses to personnel occupying the control room to within the limits of 10 CFR 50.67. The revised BVPS Unit No. 1 analysis assumes only that a 30 minute purge of the control room atmosphere occurs following radioactivity release termination. In order to ensure that sufficient electrical power is available to conduct a purge of the control room following a FHA and as a conservative measure to support the control room habitability systems, the term "recently" will not be added to the BVPS Unit No. 1 LCO applicability and action requirements for electrical power specified in LCOs 3.8.1.2, 3.8.2.2, and 3.8.2.4. Additionally, the term "recently" will not be added to the BVPS Unit No. 1 applicability and action requirements specified in Table 3.3-6 and LCO 3.7.7.1 for the control room radiation monitors and for the control room habitability systems, respectively. The basis for not adding the term "recently" to these requirements is also a conservative measure. The requirement to maintain the control room habitability systems will provide additional assurance that a sufficient level of control room ventilation equipment is available to conduct the 30 minute purge of the control room following a FHA. The term "recently" was not added to either Unit's LCOs 3.9.10 and 3.9.11, which pertain to the water level over irradiated fuel assemblies, because the radiological analysis assumes a minimum water level regardless of the amount of radioactive decay. The amount of radioactive decay and the minimum water level over irradiated fuel assemblies are two key assumptions in the FHA radiological analysis.

Consistent with the Reviewer's Note stated in TSTF 51 revision number 2, BVPS is making the commitment to the following guidelines for the assessment of systems removed from service during movement of irradiated fuel assemblies and movement of fuel assemblies over irradiated fuel assemblies:

During fuel handling, ventilation system and radiation monitors availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the fuel decays away fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay. A single normal or contingency method to promptly close containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose of the "prompt methods" mentioned above are to enable ventilation systems to draw the releases from a postulated fuel handling accident in the proper direction such that it can be treated and monitored.

Attachment E of this LAR contains the draft BVPS guidelines to implement the requirements of the above commitment for fuel handling. The guidelines are very similar to those utilized at FENOC's Perry plant. Perry's fuel handling related Technical Specifications currently contain the term "recently". The NRC approved this change by amendment number 102 dated March 11, 1999 (TAC No. M94028).

For BVPS Unit No. 1 only, LCO 3.9.4 currently requires that at least one door of the PAL be maintained closed during Core Alterations or fuel movement within containment. In addition, other containment penetrations, which provide direct access from the containment atmosphere to the outside atmosphere, must be closed. This requirement is to prevent the release of radioactive material in the event of a FHA. During fuel movement within containment, other refueling outage activities such as steam generator tube inspection and snubber inspections are normally being performed. These activities require that personnel enter and exit the containment. Since both PAL doors are normally closed to ensure that the requirements of LCO 3.9.4 are met, both air lock doors and associated opening and closing mechanisms must be cycled during each entry and exit. As a result of this frequent usage, higher wear occurs on the doors and associated closure mechanisms than would be expected if the air lock was used only for Mode 1 through 4 containment entries. Containment entry during power operation is normally infrequent.

An additional concern with requiring that PAL doors remain closed during a FHA inside containment occurs when a large number of personnel are inside containment. The PAL is the only readily accessible exit path with the containment equipment hatch closed. The emergency air lock, located in the equipment hatch, is not normally installed during refueling operations. Therefore, it may take a number of PAL door cycles to evacuate personnel from containment if a FHA were to occur. The time required for door cycling operations results in increased personnel doses for the personnel remaining in containment.

During a typical refueling outage, a number of containment isolation valves are required to be Type C leak tested in accordance with SR 4.6.1.2.a. This surveillance is conducted to ensure that containment isolation valve leakage is within acceptable limits. During this testing, the containment penetration is required to be drained of process liquid. This usually requires that both containment penetration isolation valves be opened along with the necessary vent and drain valves to permit proper draining of the piping. During this evolution a pathway from the containment atmosphere to the outside atmosphere is typically established. In addition, depending on how the air pressure in the containment penetration is bled off following completion of the leak test and the method utilized to fill and vent the containment penetration following the required testing, a pathway from the containment atmosphere to the outside atmosphere could also be established.

The proposed changes to LCO 3.9.4 applicability (for BVPS Unit No. 1 only) will permit this required testing and other required surveillance and maintenance activities to occur under the administrative controls described in Attachment E of this LAR. It is BVPS's objective to perform future refueling outages in less than 30 days. In order to support this objective, the BVPS management staff has determined that it will be necessary to allow these work activities to be conducted during movement of fuel within containment. In addition, allowing the PAL to be open will contribute to reducing the outage duration by allowing personnel quicker access to containment.

By permitting the containment equipment hatch to be open during fuel movement at both Units, activities such as steam generator sludge lancing and chemical cleaning of the steam generators will be able to be conducted without the need for special containment isolation provisions. When using this isolation arrangement, the opening of the equipment hatch for the purpose of moving equipment in and out of containment is not desirable from the standpoint of completing the outage activity in a timely manner. The removing of the equipment hatch generally requires that activity associated with the process hoses passing through the equipment hatch be temporarily terminated. The connected hoses are then isolated and removed and then reinstalled and the process flow be reestablished.

The proposed Technical Specification (TS) changes will allow both doors of the containment PAL and other containment penetrations including the containment equipment hatch to be open during fuel movement provided the fuel assemblies being moved have not been recently irradiated.

There have been several occurrences in the history of the nuclear power industry in which fuel bundles have actually been dropped in the course of fuel handling activities. In each of these instances, the actual releases from the fuel have been minimal or nonexistent (reference Nuclear Safety Analysis Center (NSAC)-129 and other subsequent plant operating event reports). This has shown that the assumptions utilized in the radiological dose calculations for a fuel handling accident are quite conservative. It has also contributed to considerations of what the actual areas of concern are during a refueling outage. It has been shown that the primary shutdown safety issue is with the potential for core damage events.

An examination of the significance of Fuel Handling Accidents was examined as part of a Grand Gulf shutdown risk study (reference NRC Meeting Summary of September 9, 1998, "Meeting To Discuss The Planned Joint Proposals On Containment Requirements To Mitigate Fuel Handling Accidents During Refueling" with several BWR/6 plants). Insights from this study show that due to the much lower potential releases from a Fuel Handling Accident than from a core damage accident (approximately 100 Curies as compared to 3 $X10^6$ Curies) the risk from a Fuel Handling Accident is very low, and is 3 orders of magnitude below the risk associated with core damage events. Although certain types of controls addressing fuel handling events remain appropriate as discussed above, these controls do not necessarily need to be addressed within the Technical Specifications.

As noted in the Commission's Statements of Consideration for the Final Rule on Technical Specification Improvements dated July 19, 1995, the Technical Specifications had grown to the point that they contained too many requirements, which led to diversion of licensee attention from the more important requirements to the extent that the excessive requirements had "resulted in an adverse but unquantifiable impact of safety." It was also noted that they had grown so large due in part to "a lack of well-defined criteria (in either the body of the rule [10 CFR 50.36] or in some other regulatory document) for what should be included in Technical Specifications."

Therefore, the Commission revised 10 CFR 50.36 to include four specific criteria on what should be included (see 10 CFR 50.36 to see the specific wording). They also noted that "If a Technical Specification provision does not meet any of the first three criteria, and if the current PRA (Probabilistic Risk Assessment) knowledge or operating experience does not identify the structure, system, or component as risk significant, the NRC staff will not preclude relocating such Technical Specifications."

Criterion 3 is the criterion that applies today to the systems, structures and components discussed in the proposed license amendment (before application of the revised Fuel Handling Accident calculations as proposed by the amendment). It reads as follows:

> "A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

This criterion was further clarified in the Commission's Final Policy Statement on Technical Specification Improvements dated July 22, 1993, which made it clear that "the primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria. It is the intent of this criterion to capture into Technical Specifications only those structures, systems and components that are part of the primary success path of a safety sequence analysis ... The primary success path ... does not include backup and diverse equipment..."

The definition of "appropriate acceptance criteria" was clearly discussed in T. E. Murley's letter dated May 9, 1988. This letter transmitted the results of the NRC Staff's review of the Owners Groups' application of the Technical Specification selection criteria and formed the basis for the issued Improved Standard Technical Specification. Enclosure Section 2.(6) states: "Accordingly, the SRP limits should be used to define the equipment in the primary success path for mitigating accidents and transients when developing the new STS."

Therefore, since the revised calculations, which credit several days of radioactive decay, do not include credit for building integrity requirements or ventilation/filtration systems as part of the primary success path, and the results of those calculations meet the SRP limits, the Technical Specification controls over such items can be relocated to procedures under licensee administrative control.

Conclusions

- Fuel Handling Accidents are not risk-significant enough to require Technical Specification controls.
- Adequate defense in depth is maintained by the requirements for water level and radioactive decay.
- The license amendment request is based on dose calculations performed assuming radioactive decay (a natural process) has occurred before the proposed changes to relax building integrity requirements take place.
- The dose calculations meet the SRP acceptance criteria without crediting the SSCs whose Technical Specifications are being revised by the proposed amendment.
- The specific Technical Specification controls addressed by the license amendment no longer meet the criteria given in 10 CFR 50.36 for inclusion within the Technical Specifications, once the fuel has undergone sufficient radioactive decay.
- Administrative controls over shutdown safety will remain in effect to control monitoring and filtration of any releases that might occur from a fuel handling accident.

DELETION OF THE TERM "LOADS" FROM THE LCO APPLICABILITY AND ACTION REQUIREMENTS

The proposed deletion of the Mode applicability of "during movement of loads over irradiated fuel assemblies" is consistent with the ISTS. For the fuel storage building, the fuel storage racks are designed to withstand the dropping of a fuel assembly and remain capable of performing their safety function to prevent a criticality event. Adequate separation is maintained between the dropped fuel assembly and the active fuel height of the stored fuel assembly which precludes interaction (reference BVPS Unit 1 LAR submittal number 202 dated November 2, 1992 (TAC No. M84673) and BVPS Unit No. 2 UFSAR section 9.1.2.3 pertaining to the spent fuel storage pool). Therefore, loads that have an impact energy which is less than or equal to the impact energy of a dropped fuel assembly (light loads) will not result in significant deformation of the fuel storage racks which will protect the stored fuel assemblies.

Crane operations involving heavy loads (greater than 2450 lbs.) over irradiated fuel stored in the spent fuel pool are prevented by administrative controls in compliance with NUREG-0612. The administrative controls are required to be implemented by NUREG-0612 and associated Generic Letter 81-07 titled "Control of Heavy Loads". As documented in the BVPS response to NUREG-0612 and the NRC SER for that response, the BVPS administrative controls for moving heavy loads provide adequate assurance that operation of the fuel building crane will not include lifting heavy loads that have the potential to damage stored fuel assemblies.

The movement of loads over irradiated fuel within the containment is addressed by the development of safe load paths consistent with the guidance provided in NUREG-0612 and the current requirement to require FHA mitigation systems when performing Core Alterations (BVPS Unit No. 1 only) and moving fuel assemblies within containment (BVPS Unit No. 2 only). These requirements provide additional assurance that FHA mitigation systems are operable when moving loads within the reactor vessel (fuel assemblies) and the potential exists for a fuel handling accident. The proposed changes to the applicable LCOs will continue to require that FHA mitigation systems be operable during movement of loads (fuel assemblies) over the reactor vessel containing recently irradiated fuel assemblies. The movement of heavy loads over the reactor core with the potential to damage fuel assemblies in the reactor core has been evaluated and safe load paths have been incorporated into the BVPS plant procedures. Changes to established safe load paths can only be made in accordance with the provisions of 10 CFR 50.59.

In addition, the proposed change to Licensing Requirements Manual (LRM) Section 7.1 titled "Crane Travel – Spent Fuel Storage Pool Building" contained in Attachment F of this LAR will further restrict the movement of loads in excess of 2450 lbs. over fuel assemblies in the fuel storage pool. A weight of greater than 2450 lbs. is the current heavy load classification limit at BVPS in accordance with NUREG-0612. The BVPS plant-specific Holtec report titled "Evaluation of Spent Fuel Assembly Drop Accidents in The Beaver Valley Power Station Reactor Core" contained in Attachment D of this LAR utilizes a total fuel assembly weight of 2500 lbs. which is 50 lbs. above the BVPS heavy load classification limit. The additional 50 lbs. was added for conservatism and to allow for additional margin should the total fuel assembly weight be increased due to changes in the fuel assembly design.

FHA ROD FAILURE ANALYSIS

NUREG 0800, Standard Review Plan (SRP) Section 15.7.4 titled "Radiological Consequences of Fuel Handling Accident." states that "The applicant should provide in the SAR conservative analyses for the number of rods assumed damaged both for the spent fuel storage area and inside containment, and the Mechanical Engineering Branch (MEB) should be requested to verify the number of rods assumed damaged. Regulatory Guide (RG) 1.183 titled "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" states the following: "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 that are utilized in the attached radiological analysis is based on the BVPS plant-specific Holtec report titled "Evaluation of Spent Fuel Assembly Drop Accidents in The Beaver Valley Power Station Reactor Core" which is contained in Attachment D of this LAR. This report is a fuel fragility analysis to determine the extent of rod breakage under various drop scenarios. This report is applicable to BVPS Unit Nos. 1 and 2. The fuel assembly is assumed to be in an irradiated state, and for conservatism, modeled to simulate the actual geometry of Westinghouse 17 x 17 Optimized Fuel Assembly (OFA), which contains rods of the smallest diameter in the family of PWR fuel assemblies, and its fuel rods are most vulnerable to drop accidents. The OFA assemblies currently are only contained in the BVPS Unit No. 1 fuel pool. These fuel assemblies were utilized as lead test assemblies a number of years ago at BVPS Unit No. 1. There is a very limited number of this type of fuel assembly in the BVPS Unit No. 1 fuel pool. There is no current plan to load this type of fuel assemblies back into the reactor vessel as part of any future core reloads. The finite-element method was used to conduct the numerical simulations for various drop events to obtain the conservative damage estimation. The analyzed drop scenarios are postulated to envelop all probabilistically significant drop events that could occur during irradiated fuel movement in the BVPS reactor core. A postulated drop of a single irradiated fuel assembly over the reactor core represents the bounding drop location. This location has the largest drop height available, based on the use of the current irradiated fuel handling procedures at BVPS, due to the reactor core barrel depth. The maximum drop height results in the maximum amount of kinetic energy available to damage the fuel assemblies. The fuel building, the fuel transfer canal and the fuel transfer tube have a smaller drop height available based on the use of the current irradiated fuel handling procedures at BVPS. In addition, only one fuel assembly is handled at a time during the transfer process between the fuel building and reactor core. Once the fuel assembly has been lifted out of the reactor core area, there are no other fuel assemblies within the target area to be damaged should the fuel assembly in transit be dropped. In the fuel pool area, the fuel racks provide additional protection against damage to additional assemblies since only one additional assembly can be struck by the dropped assembly. The target area is also limited due to the opening size of the storage location. In the reactor vessel, the target area is greatly increased due to open surface area containing fuel assemblies. In case 3 of the analysis contained in Attachment D, the event where the dropped assembly strikes the edge of a target assembly is analyzed. In this scenario, the target assembly receives the energy from the dropped assembly. This scenario bounds a scenario where more than one assembly is struck since the kinetic energy of the dropped assembly would be distributed over more than one target fuel assembly. The analysis contained in Attachment D of this LAR utilizes a total fuel assembly weight of 2500 lbs. which is 50 lbs. above the BVPS heavy load classification limit.

Therefore, the analysis contained in Attachment D of this LAR is based on the dropping of a single irradiated fuel assembly of the reactor core since this is the most limiting drop scenario at BVPS. The results of this analysis show that the number of ruptured fuel rods could be up to 137.

RADIOLOGICAL EVALUATION OF FHA

The results of the dose calculations for a FHA are presented in Attachments C-1 (BVPS Unit No. 1) and C-2 (BVPS Unit No. 2). The radiological analyses were performed based on the guidance provided in the SRP Chapter 15.0.1 and RG 1.183. The attached radiological analyses demonstrates that should a FHA occur within the containment or the fuel building that involves irradiated fuel with at least 100 hours of decay, the projected offsite doses for this event will be well within the applicable regulatory limits of 10 CFR Part 50.67 of 25 rem total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) (for any 2-hour period) and low population zone (LPZ) (for the entire period of radioactive cloud passage), and are less than the more restrictive guidance criteria in the SRP Section 15.0.1 and RG 1.183 of 6.3 rem TEDE for the 2 hour release duration. Control room personnel doses (for the duration of the accident) are less than the 10 CFR Part 50.67 limit of 5 rem TEDE. This radiological analysis is based on all airborne activity reaching the applicable building (fuel building or containment building) atmosphere, as a result of a FHA inside the applicable building, being released to the environment over a 2 hour period. The 2 hour release period is based on the guidance contained in Regulatory Guide 1.183. The radiological analyses are bounding for a FHA in the fuel building or a FHA in the containment building. These analyses use the maximum number of predicted failed fuel rods for a FHA in either building. The number of ruptured fuel rods (137) that is utilized in the attached radiological analysis is based on the BVPS plant-specific Holtec report contained in Attachment D of this LAR. The most restrictive release point atmospheric dispersion factor(s) for either building are conservatively applied to the entire release. The most restrictive of the three assumed release paths for either building is from the Primary Auxiliary Building (PAB) ventilation stack. A postulated release from the other two release paths (the containment equipment hatch or the SLCRS ventilation stack on top of the containment results in better atmospheric dispersion factor(s) than from the PAB ventilation stack.

The revised radiological analyses assumes that the radioactivity released to the fuel building or the containment building due to a FHA is then released to the environment with no credit for activity removal by filtration or plate-out. The primary success path for mitigating a FHA does not include the functioning of active containment or fuel building systems. For BVPS Unit No. 1 only, the control room habitability system (i.e., the pressurization bottles, emergency supply fans and damper isolation) is not required to mitigate the consequences of a FHA in the proposed radiological analysis in order to maintain the radiation doses to personnel occupying the control room to within the limits of 10 CFR 50.67. The revised analysis assumes that a 30 minute purge of the control room atmosphere is performed following radioactivity release termination. This post-release purge assumption has been utilized in the BVPS MSLB accident analysis which have been reviewed and been determined to be acceptable by the NRC staff. For BVPS Unit No. 2 only, the control room habitability system (i.e., the pressurization bottles, emergency supply fans and damper isolation) is also not required to mitigate the consequences of a FHA in the proposed radiological analysis in order to maintain the radiation doses to personnel occupying the control room to within the limits of 10 CFR 50.67. In addition, a post-release purge of the control room is not assumed in the revised radiological analysis.

Consistent with the requirements outlined in Attachment E of this LAR, the containment purge exhaust penetration will be in service in order to ensure that containment air will be drawn past at least one radiation monitor via the PAB ventilation release stack or via the elevated release stack on top of the containment building. The automatic isolation of the purge supply and exhaust penetrations will be defeated in order to ensure that this flow path remains available should a FHA occur within containment. If for any reason operation of the purge exhaust flow path must be discontinued during fuel movement within the containment with the equipment hatch open, the equipment hatch opening will be monitored for radioactive releases via health physics air monitoring station. In addition, if other containment penetrations are open with the purge exhaust penetration not in service, the contiguous area where the open containment penetrations are located will be verified to be exhausting via a monitored flow path or the penetrations will be isolated or fuel movement within containment will be temporarily terminated. Similarly for the fuel building, the fuel building portion of SLCRS will be operated during fuel movement within the fuel building to ensure that the fuel building air is drawn past at least one radiation monitor via the elevated release stack on top of the containment. If for any reason operation of the fuel building SLCRS flow path must be discontinued during fuel movement within the fuel building, fuel movement will be discontinued until the flow path can be reestablished.

The commitment to the previously mentioned guidelines for the assessment of systems removed from service during movement of irradiated fuel assemblies and movement of fuel assemblies over irradiated fuel assemblies, as outlined above and as specified in Attachment E of this LAR, will ensure that the radiological analyses assumption for the three release paths during a FHA remains valid. The best estimate TEDE dose for workers closing the equipment hatch following a FHA in containment is 1.1 rem TEDE based on a 20 minute stay time.

It is a normal BVPS practice, with the plant in Mode 5, to operate the non-filtered containment purge supply and exhaust system with the containment equipment hatch and PAL open. Acceptable system performance has been observed during this operating configuration. Airflow is normally observed to be into the containment through the open PAL and equipment hatch.

The following provides a review of the acceptance criteria for using alternate source term (AST) in a license amendment request as specified in the Standard Review Plan (SRP) 15.0.1:

1. 10 CFR 50.49, Environmental Qualification Of Electric Equipment Important To Safety For Nuclear Power Plants.

   Of particular concern is the qualification of equipment with regard to integrated radiation dose during normal and accident conditions. The changes proposed in this LAR do not affect normal plant operating conditions and, therefore, have no impact on the current equipment qualification status for normal plant operation. The proposed changes do affect accident conditions in the plant. However, the current environmental equipment qualification analyses utilize assumptions based on Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites." The proposed change does not include revisions to the assumptions utilized for equipment qualification. Consistent with Regulatory Position 1.3.5 of Regulatory Guide 1.183, BVPS will continue to utilize the TID 14844 assumptions for performing the required equipment qualification analyses.

2. 10 CFR 50.67, Accident Source Term.

   The proposed changes include calculations (see Attachment C of this LAR) that explain how the radiological criteria of 10 CFR 50.67 are met.

3. 10 CFR 50 Appendix A, General Design Criteria (GDC) 19, Control Room.

   The proposed changes include discussions and calculations (see Attachment C of this LAR) that explain how the control room environment is maintained within the applicable limits. The proposed changes to the technical specifications are consistent with the results of the attached calculations.

4. 10 CFR 51, Environmental Protection Regulations For Domestic Licensing And Related Regulatory Functions.

   As explained in Section G of Attachment B in this LAR, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22c(9).

5. 10 CFR 50, Appendix E, Paragraph IV.E.8, Emergency Planning And Preparedness For Production And Utilization Facilities.

   This requirement relates to the maintenance of emergency facilities in a safe, habitable condition under accident conditions. The BVPS Emergency Response Facility (ERF) is designed to accommodate continuous occupancy following a DBA. Shielding and ventilation system designs ensure occupancy of the ERF for 30 days while staying within the dose requirements of GDC 19 after a design basis LOCA. As the radiological consequences of a design basis LOCA are greater than the fuel handling accident described in this LAR, the changes proposed in this LAR do not reduce the habitability of the ERF in any way.

6. NUREG-0737 II.B.2, "Design Review Of Plant Shielding And Environmental Qualification Of Equipment For Spaces/Systems Which May Be Used In Post-accident Operations."

   The design review requirements of this item of NUREG-0737 are based on post-accident radioactivity releases equivalent to that described in Regulatory Guides 1.3 and 1.4. Regulatory Guide 1.3 pertains to Boiling Water Reactors and Regulatory Guide 1.4, titled "Assumptions Used For Evaluating The Potential Radiological Consequences Of A Loss of Coolant Accident For Pressurized Water Reactors," addresses only Loss of Coolant Accidents (LOCAs). The changes proposed in this LAR do not affect the safety analyses associated with a design basis LOCA nor do the changes in this LAR affect the plant design requirements for spaces or systems that may be used after a design basis LOCA. The existing licensing commitments in response to NUREG-0737 II.B.2 remain unaffected by the changes proposed in this LAR.

7. NUREG-0737 II.B.3, "Post Accident Sampling Capability."

   The design and operational review of sampling systems required by this item of NUREG-0737 assume accident conditions consistent with the fission product release described in Regulatory Guide 1.4 titled, "Assumptions Used For Evaluating The Potential Radiological Consequences Of A Loss of Coolant

Accident For Pressurized Water Reactors," which addresses only Loss of Coolant Accidents (LOCAs). The sampling operations in question are intended to help ascertain the degree of core damage after a LOCA. The changes proposed in this LAR do not affect the safety analyses associated with a design basis LOCA nor do the changes in this LAR affect the plant design requirements for sampling systems that may be used after a design basis LOCA. The existing licensing commitments in response to NUREG-0737 II.B.3 remain unaffected by the changes proposed in this LAR.

8. NUREG-0737 II.F.1, "Additional Accident Monitoring Instrumentation."

This item of NUREG-0737 addresses requirements for the following instrumentation:

a) Noble gas effluent monitors,
b) Iodine and particulate sampling and onsite laboratory capabilities,
c) Containment high range radiation monitor,
d) Containment pressure monitor,
e) Containment water level monitor; and
f) Containment hydrogen concentration monitor.

The changes proposed in this LAR affect the calculation method and consequences of a fuel handling accident. As a result, the changes proposed in this LAR affect specific equipment and plant conditions related to the fuel handling accident. However, the changes proposed in this LAR do not affect the requirements specified in NUREG-0737 item II.F.1 for the equipment listed above. Therefore, the changes proposed in this LAR do not affect the BVPS licensing commitments made in response to NUREG-0737 item II.F.1.

9. NUREG-0737 III.D.1.1, "Integrity Of Systems Outside Containment Likely To Contain Radioactive Material For Pressurized-Water Reactors And Boiling Water Reactors."

This NUREG-0737 item addresses requirements for a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during an accident or transient to as-low-as-practical levels. The changes proposed in this LAR do not affect the leak detection and reduction efforts related to meeting the requirements of this item of NUREG-0737. Therefore, the changes proposed in this LAR do not affect the BVPS licensing commitments made in response to NUREG-0737 item III.D.1.1.

10. NUREG-0737 III.A.1.2, Upgrade Emergency Support Facilities.

This NUREG-0737 item addresses requirements for maintaining emergency response facilities in a safe habitable condition under accident conditions by providing protection against radiation and toxic gas. The BVPS Emergency Response Facility (ERF) is designed to accommodate continuous occupancy following a DBA. Shielding and ventilation system designs ensure occupancy of the ERF for 30 days while staying within the dose requirements of GDC 19 after a design basis LOCA. A conservative evaluation of the radiation dose to an individual located in the ERF following a fuel handling accident (FHA) was performed to support this LAR. The results of this evaluation show that the radiation dose would be within 5 rem TEDE. This evaluation was based on good engineering judgment rather than a formal analysis. Following is a summary of the assumptions and methodology, and the results.

All assumptions and methodology used in the design basis analysis were used with the following two exceptions:

a) The atmospheric dispersion factor (chi-over-Q) used is that for the BVPS Unit No. 2 top of containment, to the nearest side of the ERF building. This was determined in calculation (ERS-SFL-95-012) using the NRC PAVAN code. This code was used to determine the current, licensed EAB and LPZ chi-over-Qs for both Units. For this evaluation, this is assumed to be a suitable value for any of the potential FHA release points for either Unit.

   This chi-over-Q should be conservative for any of the BVPS Unit No. 1 release points because there is considerably more distance between BVPS Unit No. 1 and the near side of the ERF building. For the other BVPS Unit No. 2 potential release points, the distance is comparable, but the release elevation is somewhat higher than that of the equipment hatch or the PAB vent. Because of this, the containment top value may not be conservative. However, other multiple conservatisms in this evaluation provide bounding results.

b) The ERF building offers some protection to lower accident radiation dose. The ERF has 2' thick concrete walls and roof which should reduce plume exposure dose to a negligible value. The outside air intake rate is limited based on an evaluation previously performed for the LOCA. Additionally, the building is provided with charcoal filters in the air recirculation system. The facility activation procedure directs that the building ventilation system be configured in the recirculation mode early in an accident. Some of the activity drawn into the building may still be retained after the plume has passed. Although this will result in protracted exposure, the net value should be less due to unmitigated plume exposure for the shorter duration.

This evaluation did not take credit for any of the ERF building protective features. It assumes that the receptor is located outside of the building at the side nearest to the release point. Thus, the dose provided in this evaluation is bounding for an individual working inside of the ERF building.

The FHA DBA computer model input file was modified considering the two items discussed above. The consequent dose was 799 mrem TEDE.

Conclusion: The ERF will remain habitable following a fuel handling DBA at either Unit.

11. NUREG-0737 III.D.3.4, Control Room Habitability Requirements.

This NUREG-0737 item addresses requirements to maintain the control room adequately protected against the effects of accidental release of toxic and radioactive gases consistent with General Design Criterion 19. The effects of the proposed changes on control room habitability are discussed in more detail in Attachments B and C of this LAR. The toxic gas protection requirements for the control room are currently specified in the BVPS LRM and are not being revised due to this LAR. In summary, the proposed changes account for control room habitability and will continue to ensure the ability to maintain the control room environment safe and within the applicable dose limits.

REVISION OF DECAY TIME

The current BVPS FHA radiological analyses are based on a decay time, resulting from the time between shutdown and movement of the first fuel assembly, of 150 hours (for BVPS Unit No. 1) and 100 hours (for BVPS Unit No. 2). The bases for the decay time specification is to ensure that sufficient time has elapsed to allow the radioactive decay of short lived fission products. The proposed decay time of

100 hours is consistent with the assumptions used in the accident analysis. The attached radiological analyses demonstrates that should a FHA occur within the containment or the fuel building that involves irradiated fuel with at least 100 hours of decay, the projected offsite doses for this event will be well within the applicable regulatory limits of 10 CFR Part 50.67 of 25 rem TEDE at the EAB (for any 2-hour period) and LPZ (for the entire period of radioactive cloud passage), and are less than the more restrictive guidance criteria in the SRP Section 15.0.1 and RG 1.183 of 6.3 rem TEDE for the 2 hour release duration. Control room personnel doses (for the duration of the accident) are less than the 10 CFR Part 50.67 limit of 5 rem TEDE. The change in decay time will result in an increase in the spent fuel pool heat load. BVPS will evaluate the effects of an increased heat load on the spent fuel pool cooling system due to conducting a core offload at 100 hours. The ability to conduct a core offload at 100 hours or (between 100 and 150 hours) will be dependent on the results of this evaluation.

Therefore, based on the above, the change in decay time specified in Specification 3/4.9.3 from 150 hours to 100 hours is acceptable.

REVISION OF FUEL BUILDING PROCESS MONITORS AND VENTILATION REQUIREMENTS

The proposed deletion of the fuel building process radiation monitors (RM-1VS-103A & B) for BVPS Unit No. 1 and (2RMF-RQ301A & B) for BVPS Unit No. 2 technical specification requirements is justified since Specification 3.9.12 will ensure that the fuel building exhaust flow is aligned to the SLCRS filter banks during plant evolutions when a FHA could occur and the filtration of the fuel building atmosphere may be required to meet the limits specified in 10 CFR 50.67. Specification 3.9.12 requires that the fuel building exhaust flow be aligned to the SLCRS filter banks during fuel movement within the fuel building. The automatic action of the BVPS Unit No. 1 fuel building process monitors to initiate filtration of the fuel building exhaust flow is, therefore, not required. The effluent monitors requirements contained in the BVPS Offsite Dose Calculation Manual (ODCM) for the SLCRS elevated release point will not be affected by this proposed deletion of the fuel building monitors. The capability to monitor this SLCRS radioactivity release point to the environment will be maintained per the ODCM requirements. The TS requirements for the fuel pool storage area monitor (RM-207) for BVPS Unit No. 1 and (2RMF-RQ202) for BVPS Unit No. 2 will also not be affected by this proposed change.

The ISTS provides guidance as to the types and bases for the radiation monitors required to remain within the TS. The radiation monitors retained in ISTS perform one of the following functional requirements that have been determined to meet the NRC's Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, 58 FR 39132, July 22, 1993 (the policy statement) criteria for inclusion in the TS. The radiation monitors retained in the TS must be:

1. Assumed in the safety analysis to provide automatic initiation of a system or component that is part of the primary success path and which functions to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier (Criterion 3),

2. Assumed in the safety analysis to provide an indication or alarm that is relied on by operators to initiate manual action that is part of the primary success path and which functions to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier (Criterion 3),

3. Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary (Criterion 1 - RCS leak detection monitors), or

4. Instrumentation that monitors Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Type A or Category I variables. Regulatory Guide 1.97 Type A variables are those indications that provide the primary information required for the control room operators to take specific manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accidents (Criterion 3). In general, Regulatory Guide 1.97 Category I (non-Type A) variables provide indications intended to assist operators in minimizing the consequences of accidents. As such, Category I (non-Type A) instrumentation has been identified as important for reducing risk to the public (Criterion 4).

The fuel building process radiation monitors do not:

1. Provide an automatic initiation function assumed in the proposed radiological analysis for any design basis accident described in BVPS Unit No. 1 UFSAR Chapter 14 or BVPS Unit No. 2 UFSAR Chapter 15. No credit for fuel building filtration during movement of non-recently irradiated fuel is assumed in the radiological analysis contained in Attachment C of this LAR. The revised requirements of LCO 3.9.12 will ensure fuel building filtration during movement of recently irradiated fuel and during movement of fuel assemblies over recently irradiated fuel assemblies without reliance on the fuel building process radiation monitors. For BVPS Unit No. 2 only, these monitors do not initiate filtration of the fuel building air exhaust flow. The fuel building is continuously exhausted to the SLCRS filter banks. For BVPS Unit No. 1 only, during plant operation in Modes 1 through 4, SLCRS flow is automatically diverted through the main filter banks on a Containment Isolation Phase A (CIA) signal. This action ensures that SLCRS exhaust flow is filtered prior to being released to the environment. SR 4.7.8.1.c.2 demonstrates this capability at least once per 18 months during plant operation. For BVPS Unit No. 2 only, during plant operation in Modes 1 through 4, SLCRS exhaust from plant areas contiguous to the containment is automatically diverted through the SLCRS filter train on a CIA signal to ensure that airflow from these plant areas is filtered prior to being released to the environment. SR 4.7.8.1.c.2 demonstrates this capability at least once per 18 months during plant operation.

2. Provide indication or alarm functions relied on by operators to take manual actions that are assumed in the safety analyses for any design basis accident described in Unit 1 UFSAR Chapter 14 or Unit 2 UFSAR Chapter 15. Manual actions for the fuel building ventilation are not assumed in the proposed radiological analysis.

3. Provide indication that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. These process monitors do not provide indication on the status of the reactor coolant pressure boundary. The level of radioactivity in the fuel building is not representative of the level of activity in the containment building where the reactor coolant pressure boundary is located, or

4. Monitor variables that have been identified as Regulatory Guide 1.97 Type A or Category I variables in the BVPS Unit 1 and Unit 2 responses to Regulatory Guide 1.97. The BVPS Unit 2 Regulatory Guide 1.97 variable Type and Category are contained in UFSAR Table 7.5-1. The BVPS Unit 1 Regulatory Guide 1.97 variable Type and Category are identified in the Unit 1 response to Generic Letter 82-33, Regulatory Guide 1.97, Revision 2, Supplemental Report, transmitted to the NRC by letter dated October 13, 1986. The fuel building process monitors are not RG 1.97 instruments.

Therefore, based on the above discussions, the proposed removal of the fuel building process radiation monitors requirements from the technical specifications are acceptable.

For a FHA in the fuel building, the radioactivity released from the fuel pool into the fuel building atmosphere is filtered by SLCRS. Currently, LCO 3.9.12 titled "Fuel Building Ventilation System – Fuel Movement" requires that the fuel building portion of SLCRS be operating and discharging through at least one train of SLCRS filters during fuel movement within the spent fuel pool. LCO 3.9.13 titled "Fuel Building Ventilation System –Fuel Storage" requires that the fuel building portion of SLCRS be operable whenever irradiated fuel is in the storage pool. These two LCOs ensure that the filtration assumptions contained in the current radiological analysis for the fuel building are met.

The proposed change to LCO 3.9.12 will require that at least one train of fuel building portion of SLCRS be operable and operating with the fuel building exhaust flow discharging through one train of the SLCRS filter banks. A single operable train of filtered SLCRS that is operating ensures that no undetected failures preventing system operation will occur, and that any active failure will be readily detected. This requirement is consistent with the required ISTS actions for an inoperable fuel building air cleanup system filter train contained in LCO 3.7.13. The essential requirements contained in Specification 3/4.9.13 will be incorporated into Specification 3/4.9.12. The remaining portions of Specification 3/4.9.13 will be deleted. Specifically, LCO 3.9.13 will be deleted. This LCO requirement is no longer necessary due to the proposed revisions to LCO 3.9.12. LCO 3.9.13 applicability is not redundant to the applicability of LCO 3.9.12. However, this LCO applicability "when irradiated fuel is in the storage pool" is not relevant to the mitigation of a FHA and can be deleted. The postulated FHA can occur only when fuel is being moved within the fuel building. If this activity is not in progress, the fuel building ventilation is not required. The proposed change in the

LCO action requirement to specify "fuel storage pool" instead of "storage pool" is editorial in nature and will make the Specification terminology consistent.

For BVPS Unit No. 1 only, SR 4.9.13 will be revised to clarify that the fuel building ventilation system is part of the SLCRS. SR 4.9.13 will be designated as SR 4.9.12.2. SR 4.9.13.a will be deleted since this requirement is contained in the current SR 4.9.13.c (see SR 4.7.8.1.a) which will be moved to SR 4.9.12. The current SR 4.9.13.b.1 requirement to verify that a high-high radiation signal diverts the fuel building to the SLCRS filter banks is not necessary due to the current requirements specified in LCO 3.9.12. LCO 3.9.12 requires that the system be aligned in this manner prior to initiating activities involving recently irradiated fuel assemblies. The proposed requirements of SR 4.9.12.1 will require verification of this flow path at least once per 12 hours.

The current SR 4.9.13 will be revised by adding an exception to item 4.7.8.1.c.2. The proposed revision of SR 4.9.13.c (for BVPS Unit No. 1) / SR 4.9.13 (for BVPS Unit No. 2) will require that the required portions of SLCRS be demonstrated operable per Specification 4.7.8.1 with exception to item 4.7.8.1.c.2. The proposed exception to SR 4.7.8.1.c.2 provides clarification that SLCRS does not have to be capable of diverting flow to the filter banks on a CIA signal. Actuation of plant equipment on a CIA signal is only required per TS Table 3.3-3 during plant operation in Modes 1 through 4. A CIA signal is generated due to increasing containment internal pressure and not due to fuel building conditions following a FHA.

The proposed revision to the current SR 4.9.12 wording is necessary to more accurately reflect the LCO requirements. The surveillance requirement to verify all building doors are closed will be clarified by specifying "the building" as "the fuel building". Also, a new footnote number 1 will be added to clarify that the fuel building doors may be opened for entry and exit. The exception is necessary to permit plant area doors to be opened for the purpose of entry and exit. During entry and exit, where the door is not held open, air infiltration into the fuel building occurs due to the building's negative pressure.

REVISION OF CONTAINMENT CLOSURE REQUIREMENTS (SPECIFICATION 3/4.9.4)

The proposed change to LCO 3.9.4.c.2 (for BVPS Unit No. 1 only) will make the LCO terminology more consistent with the associated surveillance requirement terminology. This proposed administrative change is being made to improve the presentation of this LCO requirement and is not intended to introduce a technical change.

For BVPS Unit No. 1 only, the proposed revision of the word "through" to the word "to" in SR 4.9.4.2.a is necessary to make the surveillance wording consistent with the LCO wording. The flow rate to the SLCRS is specified in the LCO. Flow rate through SLCRS is the combination of purge exhaust flow and other plant area flow rates that are serviced by SLCRS. This proposed change is consistent with the current wording contained in the BVPS Unit No. 2 SR 4.9.4.2.

The proposed changes to: SR 4.9.4.2.b to revise the reference from Specification 4.6.3.1.2 to Specification 4.9.9, the proposed deletion of SR 4.6.3.1.2.c, and the proposed addition of the requirement to SR 4.9.9 to verify that the isolation time for each system isolation valve is within limits, are administrative changes to better consolidate the purge and exhaust isolation system surveillance requirements. SR 4.6.3.1.2.c requires that each purge and exhaust containment isolation valve actuates to its isolation position on a containment purge and isolation signal. This surveillance is not relevant to the requirements of LCO 3.6.3.1. These containment isolation valves do not receive a containment Phase A or Phase B isolation signal. These valves are locked closed during plant operation in Modes 1 through 4. The automatic isolation of these valves is currently only required to mitigate a FHA within containment during cold shutdown. This surveillance requirement is also redundant to the requirements specified in SR 4.9.9. The proposed addition of the requirements to SR 4.9.9 to verify the isolation time for each isolation valve will ensure that isolation will occur within the required timeframe. Therefore, it is appropriate for the automatic isolation requirements to be specified in SR 4.9.9.

The proposed revision of SR 4.9.4.2.c (for BVPS Unit No. 1 only) will require that the required portions of SLCRS be demonstrated operable per Specification 4.7.8.1 with exception to Item 4.7.8.1.c.2. The proposed exception to SR 4.7.8.1.c.2 provides clarification that SLCRS does not have to be capable of diverting flow to the filter banks on a CIA signal. Actuation of plant equipment on a CIA signal is only required per TS Table 3.3-3 during plant operation in Modes 1 through 4. A

CIA signal is generated due to increasing containment internal pressure and not due to containment conditions due to a FHA. The proposed administrative wording change to SR 4.9.4.2.c is being made to improve the presentation of this surveillance requirement and is not intended to introduce a technical change. This change is also consistent with the current BVPS Unit No. 2 wording for SR 4.9.4.3.

For BVPS Unit No. 1 only, the proposed change to SR 4.9.4.1 to remove the wording "within 150 hours prior to the start of" and during "movement of irradiated fuel in the containment" removes unnecessary detail from the surveillance requirement. In addition, the proposed change to SR 4.9.9 to remove the wording "within 150 hours prior to the start of and" also removes unnecessary detail from this surveillance requirement. SR 4.0.4 requires that a surveillance be successfully performed and current prior to entering the Mode of applicability. Therefore, SR 4.9.4.1 and SR 4.9.9 require that these surveillances be performed within 7 days prior to entering the mode of applicability. Since the difference between 7 days and 150 hours is not significant, the existing requirement to perform these surveillances within 150 hours prior to entering the mode of applicability is essentially redundant to the requirements of SR 4.0.4.

REVISION OF CONTROL ROOM HABITABILITY REQUIREMENTS

FOR BVPS UNIT 1 only

The proposed revision to LCO 3.7.7.1.c to replace the words "or closed" with the words "or OPERABLE by being secured in a closed position with power removed" will make the operability requirements for the control room isolation dampers consistent between the two Units. If a control room isolation damper is placed in the closed position, the safety function provided by the automatic closure function has been satisfied. Therefore, this isolation damper can still be considered operable even if the automatic closure feature is not available. The automatic function of the control room isolation dampers serves only to close the dampers. Opening of the control room isolation dampers is performed by operator action. This proposed change clarifies the current LCO wording to specifically state that the control room isolation dampers can be considered operable in the closed position without the automatic closure function being available.

Specification 3/4.9.15 titled "Control Room Emergency Habitability Systems," which is applicable when both Units are in either Mode 5 or 6, is no longer necessary since the control room habitability system at either Unit will not be specifically applicable in Modes 5 or 6. The BVPS Unit No. 1 control room habitability system will be applicable when the plant is in Modes 1 through 4 and during movement of irradiated fuel assemblies or during movement of fuel assemblies over irradiated fuel assemblies. The BVPS Unit No. 2 control room habitability system will be applicable when the plant is in Modes 1 through 4 and during movement of recently irradiated fuel assemblies or during movement of fuel assemblies over recently irradiated fuel assemblies. BVPS Unit No. 2 Technical Specifications do not contain a Specification that prescribes requirements when both Units are in Modes 5 and 6. LCO 3.7.7 for BVPS Unit No. 2 contains the necessary requirements to ensure that the control room personnel are protected against radiation exposures in excess of the limits specified in 10 CFR 50.67. Similarly, LCO 3.7.7.1 for BVPS Unit No. 1 contains the necessary requirements to ensure that the control room personnel are protected against radiation exposures in excess of the limits specified in 10 CFR 50.67. Based on the fact that the BVPS control rooms are open to each other and are in the same pressure envelope, the requirements contained in each Unit's LCO for the control room habitability systems are sufficient to protect the control room operators at either Unit. Therefore, it is not necessary to state in the LCO applicability the words "at either Unit" nor to require a separate Specification to prescribe requirements when both Units are in a shutdown condition. The requirements of 10 CFR 50.36 will still be met with the proposed deletion of Specification 3/4.9.15. LCO 3.7.7.1 will continue to specify the necessary requirements to meet the assumptions for the control room habitability systems credited in the plant safety analysis.

The removal of Modes 5 and 6 applicability when both Units are in Modes 5 and 6 (currently specified in LCO 3.9.15.1) is consistent with the applicable BVPS Unit No. 1 safety analysis and the ISTS guidance. In the ISTS bases section 3.7.10 requirement to have the control room ventilation system operable in Modes 5 and 6 is based on a waste gas decay tank rupture analysis for a plant design that includes outside waste gas decay tanks. The BVPS Unit No. 1 waste gas decay tanks are located underground. The BVPS Unit No. 1 UFSAR Chapter 14 waste gas decay tank rupture analysis does not take credit for the control room ventilation system in order to limit control room personnel doses to within the limits of GDC 19 during a waste gas decay tank rupture accident.

Based on the above discussion for the deletion of the words "at either Unit" the proposed deletion of the same words for the control room isolation radiation monitors specified in Table 3.3-6 and Table 4.4-3 is acceptable.

The footnote designated by * will also be revised to clarify that emergency power for only one train of dampers is required in Modes 5, 6 and with no fuel is in the reactor pressure vessel. The proposed wording will state when emergency power is required instead of when emergency power is not required. This proposed administrative change is being made to improve the presentation of this footnote requirement and is not intended to introduce a technical change. The proposed clarification of this footnote requirement, consisting of adding the words "with no fuel in the reactor pressure vessel", is also necessary since Mode 6 is defined as the reactor head being unbolted or removed and fuel is in the vessel. Once all the fuel is removed from the reactor pressure vessel, the plant is no longer in Mode 5 or 6. However, fuel handling activities may still be occurring in the fuel building for which Specification 3.7.7.1 may be applicable. Therefore, the proposed change to this footnote will clarify that only one train of emergency power is required during this plant condition.

The proposed addition of the new footnote designated by *** will clarify that automatic actuation of the control room isolation dampers on a containment isolation phase B (CIB) signal is only required in Modes 1 through 4. Actuation of plant equipment on a CIB signal is only required per TS Table 3.3-3 during plant operation in Modes 1 through 4. A CIB signal is generated due to increasing containment internal pressure and not due to fuel building or containment conditions as a result of a FHA. Therefore, automatic closure of the control room isolation dampers on a CIB is not required to mitigate a FHA and should not be applicable during the plant conditions when only a FHA is postulated to occur.

FOR BVPS UNIT 2 only

The proposed changes to: the BVPS Unit No. 2 Applicable Modes for Table 3.3-6 item 1.c, Modes in which Surveillance required for Table 4.3-3 item 1.c and Mode Applicability for LCO 3.7.7 consists of changing from "All Modes" to "Modes 1, 2, 3 and 4, and during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies". The removal of Modes 5 and 6 from the LCO applicability is consistent with the applicable BVPS Unit No. 2 safety analysis and the ISTS guidance. In the ISTS bases section 3.7.10 requirement to have the control room ventilation system operable in Modes 5 and 6 is based on a waste gas decay tank rupture analysis for a plant design that includes outside waste gas decay tanks. The BVPS Unit No. 2 waste gas decay tanks are located underground. The BVPS Unit No. 2 UFSAR Chapter 15 waste gas decay tank rupture analysis does not take credit for the control room ventilation system in order to limit control room personnel doses to within the limits of GDC 19 during a waste gas decay tank rupture accident. The proposed revision to the applicability requirements pertaining to fuel movement will ensure that the control room radiation monitors, and the emergency air cleanup and pressurization system are operable during plant conditions where the radiological analysis does not support operation without this system and system operation may be necessary to meet the 10 CFR 50.67 limit of 5 rem TEDE during a FHA .

The proposed deletion of the requirement to suspend all operations involving positive reactivity changes from LCO 3.7.7 Mode 5 and 6 Action requirements is justified since positive reactivity changes in the reactor core due to core alterations are not postulated to result in fuel cladding integrity damage. A boron dilution event is not credible at BVPS Unit No. 2. Boron dilution during plant shutdown is controlled by LCO 3.1.2.9 titled "Isolation Of Unborated Water Sources – Shutdown. Therefore, it is acceptable to delete the required actions to suspend positive reactivity changes from LCO 3.7.7 Action requirements.

The proposed revision to the Action requirement heading of "Modes 5 and 6" is necessary to reflect the proposed LCO applicability requirements that have been previously discussed. The footnote designated by ** will also be revised to clarify that emergency power for only one train of dampers and fans is required in Modes 5, 6 and with no fuel is in the reactor pressure vessel. The proposed wording will state when emergency power is required instead of when emergency power is not required. This proposed administrative change is being made to improve the presentation of this footnote requirement and is not intended to introduce a technical change. The proposed clarification of this footnote requirement, consisting of adding the words "with no fuel in the reactor pressure vessel", is also necessary since Mode 6 is defined as the reactor head being unbolted or removed and fuel is in the vessel. Once all the fuel is removed from the reactor pressure vessel, the plant is no longer in Mode 5 or 6. However, fuel handling activities may still be occurring in the fuel building for which Specification 3.7.7 may be applicable. Therefore, the proposed change to this footnote will clarify that only one train of emergency power is required during this plant condition.

The proposed addition of the new footnote designated by *** will clarify that automatic actuation of the control room isolation dampers on a containment isolation phase B (CIB) signal is required only in Modes 1 through 4. Actuation of plant equipment on a CIB signal is only required per TS Table 3.3-3 during plant operation in Modes 1 through 4. A CIB signal is generated due to increasing containment internal pressure and not due to fuel building or containment conditions as a result of a FHA. Therefore, automatic closure of the control room isolation dampers on a CIB is not required to mitigate a FHA and should not be applicable during the plant conditions when only a FHA is postulated to occur.

MISCELLANEOUS CHANGES

The proposed change to SR 4.9.10 to remove the wording "within 2 hours prior to the start of" and "thereafter during movement of fuel assemblies or control rods" removes unnecessary detail from the surveillance requirement. SR 4.0.1 requires that surveillance requirements be met during the operational modes or other conditions specified for the individual LCO. SR 4.0.4 requires that a surveillance be successfully performed and current prior to entering the Mode of applicability. Therefore, the proposed wording for SR 4.9.10 requires that this surveillance be performed within 24 hours prior to entering the mode of applicability. The requirement to perform the surveillance within 2 hours prior to the start of, is essentially redundant to the requirements of SR 4.0.4. The requirement to perform the surveillance thereafter during the LCO applicability is unnecessary based on the requirements of SR 4.0.1. Therefore, this SR wording can be deleted.

The proposed deletion of the requirement specified in LCO 3.9.11 to restore the water level to within its limit within 4 hours is justified because the minimum water level is not required if fuel handling has been terminated. The minimum water level is an analysis assumption for a FHA. The proposed actions of LCO 3.9.11 require that fuel movement that could result in a FHA be terminated if the water level in the spent fuel pool is not within the limit. Therefore, the proposed deletion of action to restore the water level within 4 hours is acceptable. The proposed change in the LCO action requirement to specify "fuel storage pool" instead of "fuel storage areas" is necessary to make the Specification terminology consistent. This Specification uses the words "fuel storage areas" only in the action statement. The words "fuel storage pool" are utilized throughout the remaining requirements of this Specification.

The proposed change to SR 4.9.11 to remove the wording "when irradiated fuel assemblies are in the fuel storage pool" also removes unnecessary detail from the surveillance requirement. The requirement to perform the surveillance thereafter during the LCO applicability is unnecessary based on the requirements of SR 4.0.1. SR 4.0.1 requires that surveillance requirements be met during the operational modes or other conditions specified for the individual LCO. Therefore, this SR wording can be deleted.

The proposed change to SR 4.9.12 to remove the wording "within 2 hours prior to the initiation of" and "during either movement of fuel assemblies within the fuel storage pool or crane operation with loads over the fuel storage pool" removes unnecessary detail from the surveillance requirement. SR 4.0.1 requires that surveillance requirements be met during the operational modes or other conditions specified for the individual LCO. SR 4.0.4 requires that a surveillance be successfully performed and current prior to entering the Mode of applicability. Therefore, the proposed wording for SR 4.9.12 requires that this surveillance be performed within 12 hours prior to entering the mode of applicability. The requirement to perform the surveillance within "2 hours prior to the initiation of" is essentially redundant to the requirements of SR 4.0.4. The requirement to perform the surveillance thereafter during the LCO applicability is unnecessary based on the requirements of SR 4.0.1. Therefore, this SR wording can be deleted.

The editorial and format changes includes the revision of Index pages to reflect changes in Bases page numbers due to the addition of and deletion of text and Specifications, updating to current page format, the addition of new technical specification pages to accommodate the addition of text, punctuation changes, shifting of page footers, a change of the spelling of the word "airlock", adding the word "fuel" preceding the words "storage pool", and the addition of the term "assemblies" following the word "fuel." These changes are editorial in nature and are being made to improve the presentation and consistency of the Technical Specifications. The terms "Not used", "This Action is not used.", or "This Specification number is not used" will be added to replace deleted wording in order to maintain consistency through the Technical Specifications. The Bases section has been revised as necessary to reflect the changes to these Specifications, to define the term "recently", and to reflect the implementation of the AST methodology.

D. SAFETY ANALYSIS

Based on the current technical specification requirements, an environmental release due to a FHA occurring within containment is precluded by a design which automatically isolates the containment following detection of radioactivity by redundant containment purge monitors (for BVPS Unit No. 1 only). For BVPS Unit No. 2, a release due to a FHA within containment is filtered prior to being released to the environment. For a FHA occurring within the fuel building at either Unit, the release is also filtered prior to being released to the environment. The proposed amendment increases the dose at the EAB, LPZ, and to control room personnel due to a FHA occurring within the fuel building or the containment; however, the dose remains within acceptable limits. Based on the radiological analysis contained in Attachment C for a FHA without crediting the use of the filtered SLCRS or the isolation of the containment, the resultant radiological consequences of this event will be well within the applicable regulatory limits of 10 CFR Part 50.67 of 25 rem TEDE at the EAB (for any 2-hour period) and LPZ (for the entire period of radioactive cloud passage), and are less than the more restrictive guidance criteria in the SRP Section 15.0.1 and RG 1.183 of 6.3 rem TEDE for the 2 hour release duration. Control room personnel doses (for the duration of the accident) are less than the 10 CFR Part 50.67 limit of 5 rem TEDE. This radiological analysis is based on all airborne activity reaching the applicable building (fuel building or containment building) atmosphere being released to the environment over a 2 hour period. The 2 hour release period is based on the guidance contained in Regulatory Guide 1.183. Attachment E of this LAR contains draft administrative requirements to maintain the capability to close open containment penetrations following a FHA inside containment. Completion of this action will reduce the dose consequences to the environment should a FHA occur within containment by terminating the release prior to all airborne activity within the containment reaching the environment. Attachment E also contains requirements to maintain the ability to filter the fuel building exhaust air prior to being released to the environment. This capability will reduce the dose consequences to the environment should a FHA occur within the fuel building.

The proposed changes to the technical specification requirements, including changes to the LCO applicabilities and required actions, will continue to ensure that the necessary plant equipment is operable in the plant conditions where these systems are required to operate to mitigate a DBA.

The BVPS Unit No. 1 UFSAR Chapter 14 waste gas decay tank rupture analysis does not take credit for the control room ventilation system in order to limit control room personnel doses to within the limits of GDC 19 during a waste gas decay tank rupture accident.

The BVPS Unit No. 2 UFSAR Chapter 15 waste gas decay tank rupture analysis does not take credit for the control room ventilation system in order to limit control room personnel doses to within the limits of GDC 19 during a waste gas decay tank rupture accident.

LCO 3.9.10 titled "Water Level – Reactor Vessel" and LCO 3.9.11 titled "Storage Pool Water Level" will continue to ensure that at least 23 feet of water is maintained over stored/seated fuel assemblies during fuel movement. LCO 3.9.3 titled "Decay Time" will continue to ensure that irradiated fuel is not moved in the reactor pressure vessel until at least 100 hours after shutdown. These LCOs will continue to ensure that two of the key assumptions used in the radiological safety analysis for a FHA are met.

A FHA is the only event during Core Alterations that is postulated to result in fuel damage and radiological release. The accidents that are postulated to occur during Core Alterations, in addition to a FHA, are: inadvertent criticality (due to a control rod removal error or continuous control rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Therefore, the proposed change (for BVPS Unit No. 1 only) to remove the terms "Core Alterations" from LCO 3.9.4 and LCO 3.9.9 will not adversely affect the requirements to ensure that plant equipment is available during plant evolutions when a FHA can occur inside containment. Similarly, the proposed removal of the terms "or control rods" from Specification 3/4.9.10 will not adversely affect the requirements to ensure that 23 feet of water is maintained over stored fuel assemblies during plant evolutions where this water level is required to mitigate the consequences of an accident.

The radiological consequences of the Core Alteration events other than the FHA remain unchanged. These events do not result in fuel cladding integrity damage. A radioactive release to the environment is not postulated since the activity is contained in the fuel rods. Therefore, the affected containment systems and minimum water level over fuel assemblies are not required to mitigate a radioactive release to the environment due to these Core Alteration events.

The proposed deletion of the requirement (for BVPS Unit No. 2 only) to suspend all operations involving positive reactivity changes from LCO 3.7.7 Mode 5 and 6 Action requirements is safe since positive reactivity changes in the reactor core due to core alterations are not postulated to result in fuel cladding integrity damage. A boron dilution event is not credible at BVPS Unit No. 2. Boron dilution during plant shutdown is controlled by LCO 3.1.2.9 titled "Isolation Of Unborated Water Sources – Shutdown.

The proposed revision (for BVPS Unit No. 1 only) in the minimum number of the Containment Purge Exhaust Radiation Monitoring Instrumentation channels required to be operable from one to two, ensures that redundant instrument channels are available to detect and initiate isolation of the containment purge and exhaust containment penetrations during a FHA occurring inside containment.

The various administrative changes will continue to ensure that plant systems are available to support the assumptions of plant safety analysis for a FHA. Therefore, these administrative changes do not affect plant safety. The editorial and format changes do not affect plant safety.

Therefore, based on the above, this proposed amendment is considered safe.

E. NO SIGNIFICANT HAZARDS EVALUATION

The proposed amendment revises the technical specifications associated with requirements for handling irradiated fuel assemblies in the reactor containment and in the fuel building. The proposed amendment also revises the technical specifications associated with ensuring that safety analysis assumptions are met should a fuel handling accident (FHA) occur when moving irradiated fuel assemblies in the reactor containment and in the fuel building. Specifically, the radiological analysis for FHA has demonstrated that non-recently irradiated fuel does not contain sufficient fission products to require operability of accident mitigation features to meet the accident analysis assumptions. The accident mitigation features such as building integrity and engineered safety feature (ESF) ventilation systems are not required during fuel handling activities that do not involve "recently" irradiated fuel assemblies. The radiological analyses utilized to support this amendment request were performed based on the guidance provided in the NUREG 0800 titled "Standard Review Plan" (SRP) Chapter 15.0.1 and Regulatory Guide (RG) 1.183 titled "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors". The decay time specified in Technical Specification 3/4.9.3 titled "Decay Time" will be revised from 150 hours to 100 hours. The proposed amendment also includes administrative, editorial and format changes to the Specifications and Bases sections associated with handling irradiated fuel assemblies in the reactor containment and in the fuel building and with ensuring that safety analysis assumptions are met should a FHA occur.

For Beaver Valley Power Station (BVPS) Unit No. 1 only, the terms "Core Alterations" will be removed from Specifications 3/4.9.4 and 3/4.9.9. For BVPS Unit Nos. 1 and 2, the term "or control rods" will be removed from Specification 3/4.9.10.

Proposed changes to the BVPS Unit Nos. 1 and 2 Updated Final Safety Analysis Report (UFSAR) have been included in this amendment request. These UFSAR changes will revise the UFSAR description of a FHA and its radiological consequences to reflect the revised FHA analysis for each Unit.

The editorial and format changes include the revision of Index pages to reflect changes in Bases page numbers due to the addition of and deletion of text and Specifications, updating to current page format, the addition of new technical specification pages to accommodate the addition of text, punctuation changes, shifting of page footers, a change of the spelling of the word "airlock", adding the word "fuel" preceding the words "storage pool", and the addition of the term "assemblies" following the word "fuel." These changes are editorial in nature and are being made to improve the presentation and consistency of the Technical Specifications. The terms "Not used", "This Action is not used.", or "This Specification number is not used" will be added to replace deleted wording in order to maintain consistency through the Technical Specifications. The Bases section has been revised as necessary to reflect the changes to these Specifications, to define the term "recently", and to reflect the implementation of the Alternative Radiological Source Terms (AST) methodology.

The no significant hazard considerations involved with the proposed amendment have been evaluated. The evaluation focused on the three standards set forth in 10 CFR 50.92(c), as quoted below:

> The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment 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 following evaluation is provided for the no significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed amendment involves changes to accident mitigation system requirements. These systems are related to controlling the release of radioactivity to the environment and are not considered to be accident initiators to any previously analyzed accident.

   The implementation of the AST methodology does not affect the probability of an accident previously evaluated. The AST methodology is utilized to evaluate the consequences of a FHA.

   Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   The proposed amendment increases the dose at the exclusion area boundary (EAB), low population zone (LPZ), and to the control room personnel due to a FHA. However, the dose remains within acceptable limits. Based on the revised radiological analysis for a FHA which does not credit filtration of the radioactive release or the isolation of the containment, the resultant radiological consequences of this event at the EAB and outer boundary of the LPZ are well within the 10 CFR Part 50.67 limits as defined by acceptance criteria in the Standard Review Plan (SRP) Section 15.0.1. Control room personnel doses (for the duration of the accident) remain less than the 10 CFR Part 50.67 limit of 5 rem total effective dose equivalent (TEDE).

   The proposed revision of the decay time from 150 hours to 100 hours is consistent with the assumptions used in the FHA accident analysis. The revised radiological analyses demonstrates that should a FHA occur within the containment or the fuel building that involves irradiated fuel with at least 100 hours of decay, the projected offsite doses for this event will be well within the applicable regulatory limits of 10 CFR Part 50.67 of 25 rem TEDE at the EAB (for any 2-hour period) and LPZ (for the entire period of radioactive cloud passage), and are less than the more restrictive guidance criteria in the SRP Section 15.0.1 and RG 1.183 of 6.3 rem TEDE for the 2 hour release duration. Control room personnel doses (for the duration of the accident) are less than the 10 CFR Part 50.67 limit of 5 rem TEDE.

Limiting Condition for Operation (LCO) 3.9.10 titled "Water Level – Reactor Vessel" and LCO 3.9.11 titled "Storage Pool Water Level" will continue to ensure that at least 23 feet of water is maintained over the stored/seated fuel assemblies during fuel movement. LCO 3.9.3 titled "Decay Time" will continue to ensure that irradiated fuel is not moved in the reactor pressure vessel until at least 100 hours after shutdown which is consistent with the revised FHA radiological analysis. These LCOs will continue to ensure that two of the key assumptions used in the radiological safety analysis are met.

The radiological consequences of the Core Alteration events other than the FHA remain unchanged. These events do not result in fuel cladding integrity damage. A radioactive release to the environment is not postulated since the activity is contained in the fuel rods. Therefore, the affected containment systems are not required to mitigate a radioactive release to the environment due to a Core Alteration event.

The proposed deletion of the requirement (for BVPS Unit No. 2 only) to suspend all operations involving positive reactivity changes from LCO 3.7.7 Mode 5 and 6 Action requirements does not affect the radiological consequences to control room personnel due to this event. Positive reactivity changes in the reactor core due to core alterations are not postulated to result in fuel cladding integrity damage. A boron dilution event is not credible at BVPS Unit No. 2. Boron dilution during plant shutdown is controlled by LCO 3.1.2.9 titled "Isolation Of Unborated Water Sources – Shutdown.

The BVPS Unit No. 1 UFSAR Chapter 14 waste gas decay tank rupture analysis does not take credit for the control room ventilation system in order to limit control room personnel doses to within the limits of 10 CFR 50

Appendix A General Design Criterion (GDC) 19 during a waste gas decay tank rupture accident.

The BVPS Unit No. 2 UFSAR Chapter 15 waste gas decay tank rupture analysis does not take credit for the control room ventilation system in order to limit control room personnel doses to within the limits of GDC 19 during a waste gas decay tank rupture accident.

The proposed revision in the minimum number of the Containment Purge Exhaust Radiation Monitoring Instrumentation channels required to be operable from one to two (for BVPS Unit No. 1 only), ensures that redundant instrument channels are available to detect and initiate isolation of the containment purge and exhaust containment penetrations during a FHA inside containment involving recently irradiated fuel assemblies.

The proposed amendment does not alter the manner in which fuel assemblies are handled or core alterations are performed. The proposed amendment does not alter the manner in which heavy loads are controlled at the Beaver Valley Power Station.

The proposed changes to the technical specification requirements, including changes to the LCO applicabilities and required actions, will continue to ensure that the necessary plant equipment is operable in the plant conditions where these systems are required to operate to mitigate a Design Basis Accident (DBA).

The proposed administrative, editorial, and format changes do not affect plant safety. The Bases section has been revised as necessary to reflect the changes to these Specifications.

Therefore, the proposed amendment does not significantly increase the consequences of any previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed amendment affects a previously evaluated accident; e.g., FHA. The proposed amendment does not represent a significant change in the configuration or operation of the plant. The proposed amendment takes credit for the normal decay of irradiated fuel rather than crediting active mitigative systems (e.g., ventilation and filtration systems). Since radioactive decay is a natural phenomenon, it has a reliability of 100 percent in reducing the radiological releases from irradiated fuel assemblies. In addition, the water level that covers irradiated fuel assemblies is another natural method that provides an adequate barrier to a significant radiological release. The proposed amendment does not adversely impact Technical Specification requirements for systems needed to prevent or mitigate other Core Alteration events.

The proposed amendment does not alter the manner in which fuel assemblies are handled or core alterations are performed. The proposed amendment does not alter the manner in which heavy loads are controlled at the Beaver Valley Power Station.

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

3. Does the change involve a significant reduction in a margin of safety?

The proposed amendment increases the dose at the EAB, LPZ, and to the control room personnel due to a FHA occurring within containment or the fuel handling building; however, the dose remains within acceptable limits as defined by RG 1.183, SRP 15.0.1 and 10 CFR 50.67. The revised radiological analysis based on the proposed amendment and the use of the AST methodology, demonstrates that during a FHA, the projected offsite doses will be well within the applicable limits of 10 CFR Part 50.67 of 25 rem TEDE at the EAB (for any 2-hour period) and LPZ (for the entire period of radioactive cloud passage). The projected offsite doses are also less than the more restrictive regulatory limit specified in RG 1.183 and the guidance criteria stated in the SRP Section 15.0.1 of 6.3 rem TEDE for the 2 hour release duration. Control room personnel doses (for the duration of the accident) are less than the 10 CFR Part 50.67 limit of 5 rem TEDE. This radiological analysis is based on all airborne activity reaching the containment or fuel building atmosphere, as a result of a FHA within the containment or fuel handling building, being released to the environment over a 2 hour period. The 2 hour release period is based on the guidance contained in RG 1.183. This amendment request contains administrative requirements to maintain the capability to close open containment penetrations following a FHA inside containment. Completion of this action will reduce the dose consequences to the environment should a FHA occur within containment by terminating the release prior to all airborne activity being released from the containment. This amendment request also requires the capability of directing the fuel building exhaust air to filtered SLCRS prior to being released to the environment. This capability will reduce the dose consequences to the environment should a FHA occur within the fuel building.

The proposed revision of the decay time from 150 hours to 100 hours is consistent with the assumptions used in the FHA accident analysis. The revised radiological analyses demonstrates that should a FHA occur within the containment or the fuel building that involves irradiated fuel with at least 100 hours of decay, the projected offsite doses for this event will be well within the applicable regulatory limits of 10 CFR Part 50.67 of 25 rem TEDE at the EAB (for any 2-hour period) and LPZ (for the entire period of radioactive cloud passage), and are less than the more restrictive guidance criteria in the SRP Section 15.0.1 and RG 1.183 of 6.3 rem TEDE for the 2 hour release duration. Control room personnel doses (for the duration of the accident) are less than the 10 CFR Part 50.67 limit of 5 rem TEDE.

Therefore, the margin of safety is not significantly reduced based on meeting the applicable regulatory limits for a FHA which are conservatively set below the 10 CFR 50.67 limits for the EAB and LPZ. With respect to the control room personnel doses, the margin of safety is not significantly reduced based on the doses remaining less that the 10 CFR 50.67 limit of 5 rem TEDE.

The margin of safety for Core Alteration events other than the FHA is not significantly reduced due to this proposed amendment. The proposed amendment does not impact Technical Specification requirements for systems needed to prevent or mitigate such Core Alteration events. These events do not result in fuel cladding integrity damage. Therefore, a radioactive release to the environment is not postulated since the activity is contained in the fuel rods.

The proposed deletion of the requirement (for BVPS Unit No. 2 only) to suspend all operations involving positive reactivity changes from LCO 3.7.7 Mode 5 and 6 Action requirements does not affect the radiological consequences to control room personnel due to this event. Positive reactivity changes in the reactor core due to core alterations are not postulated to result in fuel cladding integrity damage. A boron dilution event is not credible at BVPS Unit No. 2. Boron dilution during plant shutdown is controlled by LCO 3.1.2.9 titled "Isolation Of Unborated Water Sources – Shutdown.

The proposed amendment does not alter the manner in which fuel assemblies are handled or core alterations are performed. The proposed amendment does not alter the manner in which heavy loads are controlled at the Beaver Valley Power Station.

The proposed revision in the minimum number of the Containment Purge Exhaust Radiation Monitoring Instrumentation channels required to be operable from one to two (for BVPS Unit No. 1 only), ensures that redundant instrument channels are available to detect and initiate isolation of the containment purge and exhaust containment penetrations during a FHA inside containment involving recently irradiated fuel assemblies.

The BVPS Unit No. 1 UFSAR Chapter 14 waste gas decay tank rupture analysis does not take credit for the control room ventilation system in order to limit control room personnel doses to within the limits of GDC 19 during a waste gas decay tank rupture accident.

The BVPS Unit No. 2 UFSAR Chapter 15 waste gas decay tank rupture analysis does not take credit for the control room ventilation system in order to limit control room personnel doses to within the limits of GDC 19 during a waste gas decay tank rupture accident.

The proposed changes to the technical specification requirements, including changes to the LCO applicabilities and required actions, will continue to ensure that the necessary plant equipment is operable in the plant conditions where these systems are required to operate to mitigate a DBA.

The proposed administrative, editorial, and format changes do not affect plant safety. The Bases section has been revised as necessary to reflect the changes to these Specifications.

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

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided guidance concerning the application of standards in 10 CFR 50.92 by providing certain examples (March 6, 1986 51FR7751) of amendments that are considered not likely to involve a significant hazards consideration. The proposed amendment is similar to example (vi) stated in the March 6, 1986 Federal Register Notice, in that this proposed change results in an increase in the consequences of a previously analyzed accident. However, the results of this change are within the acceptance criteria stated in SRP Section 15.0.1 of 6.3 rem TEDE. Control room personnel doses are less than the 10 CFR Part 50.67 limit of 5 rem TEDE.

Based on the considerations expressed in this application for license amendment, it is concluded that the activities associated with this license amendment request satisfy the requirements of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL CONSIDERATION

This license amendment request changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. It has been determined that this license amendment request involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. This license amendment request may change requirements with respect to installation or use of a facility component located within the restricted area or change an inspection or surveillance requirement; however, the category of this licensing action does not individually or cumulatively have a significant effect on the human environment. Accordingly, this license amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this license amendment request.

H. UFSAR CHANGES

Draft UFSAR changes are provided in Attachment G.

ATTACHMENT C-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 219

Safety Analysis of the Radiological Consequences of a Fuel Handling DBA at BVPS Unit 1, Control Room, EAB and LPZ Doses

RTL: A9.621A
Form: RE 1.103-1

Beaver Valley Power Station

Health Physics Section

REVISION	[redacted]	2	3	

Subject		
Safety Analysis of the Radiological Consequences of a Fuel Handling DBA at BVPS Unit 1, Control Room, EAB and LPZ Doses	**ERS-JTL-99-009**	PAGE 1 OF **26**

Reference

HPM RP/RIP_______ EPP_______ T/S_______ EM_______ DCP_______

Review Category

☒ RSC Required ☐ RSC Not Required ☐ 10 CFR 50.59 Required

Unit 1 ☒ Unit 2 ☐

Purpose

This calculation package documents an analysis of the postulated dose in the common control room, at the Exclusion Area Boundary (EAB), and at the Low Population Zone (LPZ) following a Fuel Handling DBA at BVPS Unit 1. This revision follows the guidance provided in USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

NOTE: This calculation package documents the evaluation described above. This package does not, alone, provide for any revision in a structure, system or component; nor changes in procedures, tests, or experiments described in the plant licensing basis. The data and/or conclusions of this package shall not be extended to other procedures without explicit concurrence from Health Physics.

Rev.	Signatures	Description
	by ___ date chk ___ date app ___ date	
3	by ___ date chk ___ date app ___ date	
2	by John T. Lebda 10-18-00 date chk 10/18/00 date app FOR M+4, RSC (12-00) 10/27/00 date	FHA assumed at 100 hours after reactor shut down. Used maximum number of damaged fuel assembly rods (137) that could result, assuming the worst-case conditions that may be encountered in either the fuel building or the reactor containment building at Unit 1. Uses the current analysis assumptions and input parameters to the extent that they are consistent with those specified in Regulatory Guide 1.183 - changed pool water iodine DF, activity fraction in-gap, submersion dose conversion factors and calculated TEDE.
1	by John T. Lebda date chk ___ date app ___ date	See previous revisions for associated signature pages. This is a revised cover sheet,
0	by John T. Lebda date chk ___ date app ___ date	**Checklist:** ☒ Purpose ☒ Assumptions ☒ Methodology ☒ Input Data ☒ Results ☒ References **Attachments:** ☒ Data Sheets ☒ Illustrations ☒ Printouts ☐ Code Listings

☒ BV RECORDS CENTER	☐ Supv, Rad Ops-1	☒ Author: J. Lebda
☒ CALCULATION FILE	☐ Supv, Rad Ops-2	☒ NS&L, A. Dometrovich
☒ MGR, Health Physics	☐ Supv, Effl & Rad Waste	☒ Mark Duranko
☒ Supv, Rad Eng & Health	☐ Training Section	☐ ________

DISCUSSION

General

This calculation determines control room operator, exclusion area boundary (EAB) and low population zone (LPZ) radiation doses following a fuel handling design basis accident (FHA DBA) at Beaver Valley Unit 1. The plant parameters and assumptions used herein are consistent with the plant design basis as revised in accordance with Regulatory Guide 1.183[1], and they provide results that define the upper bound of the accident dose consequences. As a design basis calculation, this is not intended to show what the expected doses would be, but rather what they might be if the plant is operated at "worst case" design limits (operating at the NRC License limits) and using "worst case" accident conditions. The combination of circumstances that would result in the associated maximum dose is unlikely to occur.

This revision of the FHA was requested by the Licensing Section. It is performed to demonstrate that the design case accident dose will remain within the regulatory dose limits and criteria with certain changes made to plant operation and configuration. Additionally, this analysis represents Beaver Valley Unit 1's first use of the alternative source term as provided in the recently issued Regulatory Guide 1.183[1]. As such, the design basis of the facility is changed, and NRC review is required prior to implementation of any of the configuration/operational changes supported by the analysis. The facility changes made herein are specific to the FHA scenario, and affect the activity available for release and pre-release treatment for this accident only. The characteristics of plant configuration and/or operation associated with the other design basis accidents are not changed. Thus the validity of each of the other design basis accidents as described in the facility UFSAR is unaffected and a clear, consistent and logical design basis is maintained. Refer to the Input Data and Assumptions section of this calculation package for additional information concerning the analysis methodology and assumption changes.

Fuel Handling Accident

This DBA is described in NRC Regulatory Guide 1.183[1] and NUREG-0800 Chapter 15, Section 15.0.1[2]. The accident occurs while moving a fuel assembly in either the fuel building fuel storage pool or in the reactor building containment cavity or transfer canal. The assembly is dropped, resulting in rupture of 137 fuel rods (in the dropped assembly plus other assemblies that may be struck) and release of radioactive iodine and noble gas into the pool water. The extent of damage has been determined by performing an analysis[3] using the limiting drop conditions and considering the weight of the 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 assemblies has been considered.

As the iodine percolates to the pool surface much of it will be "scrubbed out" and remain in the water. This reduces the amount of iodine that is released from the damaged rods by a factor of 200 before it becomes airborne within the building. The noble gas activity released from the damaged rods is all released to the building with no removal by pool water scrubbing. After becoming airborne, the radioactivity is released to the environment assuming a constant air flow rate (exponential activity removal rate). The analysis model uses a conservatively calculated release rate constant that results in 99.9999% of the activity being released to the environment in the two hours immediately following the accident. Because the accident conditions may include having any of the reactor building containment penetrations open (including the equipment hatch or personnel airlock), and the release may be via any one or a combination of penetrations, the most restrictive release point atmospheric dispersion factor(s) are conservatively applied to the entire release. For this analysis to remain valid, the radioactivity release must be via one of these three points. Additionally, this analysis conservatively does not take credit for any pre-release filtration or iodine plate-out.

Calculation History

SWEC RP-11700-043-0, 1, -2, -3 (1977)[4]
Calculated site boundary doses. Superseded by SWEC RP-11700-131-1

SWEC RP-11700-131-0, -1(1977)[5]
Calculated site boundary doses. Revision 1 changed the decay time (time between reactor subcritical to the accident) from 100 to 150 hours. Superseded by HOTEC calculation.

SWEC 12241 UR(B)-447-0 (1987)[6]
Calculated control room operator dose for the combined (Unit 1 + Unit 2) control room. Superseded by SWEC 12241 UR(B)-456-0.

SWEC 12241 UR(B)-456-0, -1 (1988)[7]
Calculated control room operator dose for the combined (Unit 1 + Unit 2) control room. Increased fuel building ventilation flow rate from 2000 cfm to 3000 cfm (rev. 1); analyzed two cases, using SLCRS iodine filter efficiency of 95% and 90% (90% and 85% in rev. 1). Updated control room atmospheric dispersion factors. Analyzed with control room isolation. Superseded by ERS-SFL-92-025-0.

ERS-SFL-90-002-0 (1990)[8]
Special calculation to assess effect of using a 60 day decay time and delaying SLCRS diversion for 20 minutes.

ERS-SFL-92-025-0 (1992)[9]
Calculated control room operator dose for the combined (Unit 1 + Unit 2) control room. Analyzed with no control room isolation/filtration. Changed SLCRS iodine filter efficiency to 95%. Updated control room atmospheric dispersion factors to current, approved values. Increased iodine 131 gap fraction from 0.10 to 0.12. Used revised decayed core radionuclide inventory calculated using ORIGEN2.
***** Calculation of record for control room operator dose *****

HOLTEC Calculation[10]
Calculated environmental (EAB and LPZ) dose. Updates decayed core radionuclide inventory to that calculated using revised core parameters (extended burn-up) and ORIGEN2. Changed SLCRS iodine filter efficiency to 95%. Increased iodine 131 gap fraction from 0.10 to 0.12.
***** Calculation of record for environmental dose *****

ERS-JTL-99-009-0 (1999)[11]
This revision was prompted by References 12 and 13, and the associated extent of condition review and corrective actions specified therein. Calculated control room operator dose for the combined (Unit 1 + Unit 2) control room, and offsite dose at the EAB and LPZ. Updated EAB and LPZ atmospheric dispersion factors (chi/Q) to current, approved values. Modeled the release as a "puff" for control room dose analysis (consistent with the EAB/LPZ analysis). Updated decayed core radionuclide inventory to that calculated using revised core parameters (extended burn-up) and ORIGENS[14]. Also, the core inventory value used is determined by selecting the maximum activity from a range of core enrichments for each radioisotope. Used updated ICRP 26/30 dose quantities and dose conversion factors[15,16]. This calculation has received internal review; however, it has not been reviewed and accepted by the NRC. Consequently it is not a part of the Unit 1 design basis.

ERS-JTL-99-009-1 (2000)[17]
Included added detail on the assumptions regarding iodine chemical composition and associated charcoal filter removal efficiencies. Changed the total SLCRS iodine removal efficiency from 95% to 90%. This calculation has received internal review; however, it has not been reviewed and accepted by the NRC. Consequently it is not a part of the Unit 1 design basis.

THIS CALCULATION - ERS-JTL-99-009-2 (2000)
Assumes that the FHA may occur as early as 100 hours after reactor shut down. Uses the maximum number of damaged fuel assembly rods (137) that could result, assuming the worst-case conditions that may be encountered in either the fuel building or the reactor containment building at Unit 1. The validity of this radiological consequence analysis is contingent upon NRC review and acceptance of the engineering analysis that determined the maximum number of rods damaged. Uses the current analysis assumptions and input parameters to the extent that they are consistent with those specified in Regulatory Guide 1.183[1]. Changes made to conform with this Guide affect pool water iodine DF, activity fraction in-gap, reference source for submersion dose conversion factors and the dose quantity that is calculated. All changes are detailed in the Input Data and Assumptions section of this analysis.

METHODOLOGY

Overall Methodology

The methodology used in this analysis is similar to that used in previous analyses. This revision uses version 1.0b of the TRAILS_PC (for Transport of Radioactive mAterial In Linear Systems), PC version documented in Reference 18. This version is similar to the VAX-based version used in the current calculation of record for control room dose, and has been verified to produce like results. This newer version has the capability to model progeny in-growth, a feature used in this analysis. Also, the submersion dose conversion factors from DOE/EH-0700[16] (used in previous revisions of the FHA analysis) have been replaced with those from Federal Guidance Report No. 12[19]. This "b" version of TRAILS_PC has been designated as TRAILS12 to readily identify this change to the user. More details are provided with the TRAILS_PC input file in Attachment 3.

Fuel Handling Accident Modeling

The FHA model is relatively simple and is depicted in Attachment 1.

INPUT DATA AND ASSUMPTIONS

1.0 Assumptions

1.1 This analysis of the Unit 1 FHA is based on guidance provided in NUREG-0800, Chapter 15.0.1[2] and USN. .C Regulatory Guide 1.183[1].

1.2 Minimum 100 hours between reactor shutdown (subcriticality) to the accident release. [Assumption]

This value is decreased from the current Technical Specification[20] value of 150 hours with the intent of supporting a change to this requirement.

1.3 Radioactivity release to the building occurs instantaneously, followed by a release to the environment for a two hour period. [1]

To model a constant air flow, a release rate constant which results in 99.9999% of the total radioactivity released in two hours is used. This approach is conservative particularly for the control room operator and EAB 0-2 hour doses. Release rate is important where control room radiation monitor response is needed, and the alarm time will be calculated.

This calculation takes no credit for this protective feature; however, the monitor alarm and actuation of the control room habitability system are expected to occur. Regulatory Guide 1.183[1] specifies that the EAB and LPZ doses be calculated for the two hour period that results in the highest dose. Because, per the Guidance, effectively all of the activity from the fuel handling accident Is released to the environment in the first two hours, only this time period need be considered.

1.4 The radioactivity released to the building is then released to the environment with no credit for activity removal by filtration or plate-out. [Assumption]

While some of the release pathways may be filtered, this will not be credited in this analysis. Assuming no filtration is a conservative assumption.

2.0. Input Data

2.1 Core activity in gap: [1]

I-131	=	8%
Other iodines	=	5%
Kr-85	=	10%
Other noble gases	=	5%

2.2 Radial peaking factor = 1.65 [References 4 - 11]

For events that do not involve the entire core, Regulatory Guide 1.183[1] provides "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". 1.65 is the value used in previous analyses and is slightly higher than the facility-specific value. Continued used of this value is slightly conservative.

2.3 Number of assemblies in core = 157 [21, 22]

2.4 Number of ruptured fuel rods = 137 [3]

2.5 Number of rods in assembly = 264 [21, 22]

2.6 Fuel pool iodine DF = 200 [1]

2.7 Core Inventory at T = 100 hours, and release to the building: [23, 24]

Calculated in Attachment 2

Regulatory Guide 1.183[1] Appendix A provides "Radionuclides that should be considered include xenons, kryptons, halogens, cesiums and rubidiums". Additionally, Section 4.1.1 provides in part, for the EAB and beyond, "The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity".

Although this statement is not specifically repeated for the control room, Section 4.2.2 provides "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". Considering this guidance, the source term from the previous revision of this calculation (includes kryptons, xenons and iodines) is deemed appropriate for continued use based on the following.

Cesium and rubidium are particulates, therefore the activity released from the fuel will be retained in the pool water. However, cesium may be produced in the building and the environment as progeny of the I-135 – Xe-135 – Cs-135 decay chain. Because of the 100 hour pre-accident decay period, the 135 decay chain is the only one that has precursor isotopes of sufficiently long half-life to contribute to post-release cesium production. Cs-135 has a very long half-life and it's precursors have short half-lives. Consequently, the total activity of Cs-135 that could be produced from the released I-135 and Xe-135 is minute, and is not considered to be significant with regard to either submersion or internal dose. Post-release rubidium production will not contribute to dose. Kr-85 and Kr-85m are the only precursor isotopes that remain after the 100 hour pre-accident decay period, and these decay to stable Rb-85.

The halogen, Br-82, will be present in the fuel clad gap in small quantities even after the 100 hour pre-release decay period. The dose contribution from this isotope was evaluated and, assuming that pool scrubbing will occur similar to that for iodine, this contribution is not significant, i.e., will not change the dose values calculated in this analysis.

2.8 Pool release iodine composition = *N/A for this analysis* [N/A]

Regulatory Guide 1.183[1] provides specific fractions for the iodine release chemical form. These become important when determining removal by filtration, an assumption NOT made in this analysis.

2.9 Two hour duration release rate constant (λ) = 1.92E-03 s^{-1} [Assumption]

For a two hour release duration, 99.9999% of the radioactivity is assumed to be released in the two hours.

$$0.000001 = e^{-\lambda t}$$

$$\ln(0.000001) = \lambda t$$

$$13.82/t = \lambda$$

where $t = 7200$ seconds, $\lambda = 1.92\text{E-}03\ s^{-1}$

2.10 Control room air intake and exhaust flow rate: [21, 22, 25]

T = 0 – 30 minutes____________500 cfm intake & exhaust
(Unfiltered, normal mode)

T = 30 – 120 minutes__________0 cfm intake & exhaust
(Emergency pressurization mode)

T = 120 – 150 minutes_________16,900 cfm intake & exhaust
(Unfiltered, purge mode)

T = 150 min – 30 days_________500 cfm intake & exhaust
(Unfiltered, normal mode)

The control room ventilation system parameters are selected to provide bounding analysis results and to permit maximum operational flexibility. Preliminary analysis results demonstrated that the accident dose would not be acceptable with the control room ventilation system remaining in the normal configuration. Some aspect(s) of this protective system would need considered. The only mitigation feature that is credited to reduce control room operator dose is system operation in the purge mode for 30 minutes after the end of the radioactivity release to the environment. While the actual timing and flow rates for the other system parameters may vary from the assumptions given above, sensitivity analyses have been performed which demonstrate that such variations would act to decrease the analysis dose.

The first time period is the duration of the significant radioactivity release. Because of the exponential release model, little radioactivity is released after this period. By assuming the normal, unfiltered mode of operation, radioactivity drawn into the control room is maximized. This is a bounding assumption, because control room isolation (actuation of the emergency habitability system) will occur before 30 minutes. This will be accomplished automatically by the radiation monitor alarm, assuming the radioactivity release is high enough. (As noted below, radioactivity from as little as 50 rods broken will cause an alarm). Sensitivity analyses (on file) were performed which demonstrate that decreasing the isolation time will decrease the operator dose.

The second time period is the remainder of the two hour radioactivity release period specified by Regulatory Guide 1.183[1]. This period is conservative because actions by plant personnel are expected to isolate release pathways within two hours of the accident. At 30 minutes the exponential release modeling has maximized the radioactivity in the control room. Although the air outside of the control room may still contain residual radioactivity, sensitivity analyses (on file) demonstrate that any air exchange with between the environment and the control room will reduce the control room air radioactivity concentration. Assuming 10 cfm unfiltered inleakage was insufficient to change the dose, but the radioactivity content showed a slight decrease.

When this was increased to 310 cfm, the dose decreased. Although the control room envelope may be pressurized by clean, bottled air (600 cfm minimum) for 30 – 90 minutes, and by filtered outside air (600 cfm minimum) for the remainder of the radioactivity release period, the affect of this air exchange is not credited in this analysis.

The third time period is important with regard to this analysis. The post release purge will remove most of the airborne radioactivity that has accumulated in the control room. Step 10 of 1/2-OM-44A.4A.A Part B[26] (entered from Irradiated Fuel Damage Abnormal Operating Procedure 1OM-53C.4.1.49.1[27]) directs the operator to purge the control room with at least 16,900 cfm for a minimum of 30 minutes upon Emergency Director determination that the event that resulted in control room isolation is terminated. (16,000 cfm is the lower value for the Unit 2 purge system, and is a conservative value in this analysis.). Additional dose reduction will result if the radioactivity release is terminated earlier and this action is performed sooner. This post-release purge assumption has been used in other BVPS accident analyses[28,29] which have been reviewed and accepted by the NRC.

The fourth and final time period is the return to normal system operation. If the purge duration is extended beyond 30 minutes, the dose will be further reduced. Assuming the minimum purge duration in the analysis is conservative.

Sensitivity analyses (on file) were performed to answer the question "Is it possible to reduce the amount of fuel damage to the point where the control room radiation monitor would not reach the control room habitability system automatic actuation setpoint (process safety limit of 1 mR/hr[30,31]), and still have enough radioactivity released to exceed the control room operator dose resulting from the case described above?" With 50 rods damaged, the setpoint is still reached; however, the control room operator dose is less that in the case described above. Therefore, the control room ventilation and emergency habitability system parameters used in this analysis represent the bounding accident conditions.

2.11 Control room volume = 1.73E+05 ft^3 [21, 22, 25]

2.12 Control room occupancy factors: [1]

0 to 24 hours = 1.0
24 hours to 4 days = 0.6
4 to 30 days = 0.4

2.13 Control room operator breathing rate = 3.5E-04 m^3/s [1]

2.14. Offsite breathing rates (0 to 8 hours) = 3.5E-04 m^3/s [1]

2.15 Atmospheric dispersion factors: [32,33,34,35,36]

	EAB (s/m^3)	LPZ (s/m^3)	Control room (s/m^3)
0 – 2 hours	1.04E-03		
0 – 8 hours		6.04E-05	4.30E-03 Primary Auxiliary Building

Note: References 32 and 33 are related to Unit 2 submittals. Revised atmospheric dispersion factors were approved by the NRC for both Units 1 and 2 therein. Because this release is conservatively modeled as a having a two hour duration, atmospheric dispersion factors beyond this time are unnecessary.

Of the three possible release pathways (equipment hatch and SLCRS and stacks), the most restrictive is that for the primary auxiliary building. If any portion of the radioactivity is released via the other two pathways, the control room dose would be reduced. This is not an issue with regard to offsite atmospheric dispersion values as these are not specific to the individual release points.

RESULTS AND CONCLUSIONS

The assumptions and data given above were used to build the TRAILS_PC (v1.0b) input file for the control room, LPZ and the EAB dose calculation. The input and output files are given in Attachment 3. The control room operator, EAB and LPZ calculated doses are summarized below.

	TEDE (rem)
Control room	2.2E+00
EAB	1.7E+00
LPZ	9.5E-02

Offsite doses are within the applicable regulatory limit of 10 CFR 50.67[37] of 25 rem TEDE at the EAB (for any 2-hour period) and LPZ (for the entire period of radioactive cloud passage), and are less than the more restrictive guidance criteria in the NUREG-0800 Section 15.0.1[2] and Regulatory Guide 1.183[1] of 6.3 rem TEDE for the 2 hour release duration. Control room operator doses (for the duration of the accident) are less than the 10 CFR 50.67[37] limit of 5 rem TEDE.

REFERENCES

1. USNRC Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, 2000

2. USNRC Standard Review Plan (NUREG-0800) SRP 15.0.1, Radiological Consequence Analyses Using Alternative Source Terms, 2000

3. HOLTEC Report No. HI-992343, Project No. 90844, Evaluation of Spent Fuel Assembly Drop Accidents in the Beaver Valley Power Station Reactor Core, 2000

4. SWEC Calculation Package RP-11700-043-3, Fuel Handling Accident Site Doses Based on BV1 SER X/Q = 1.3E-03 sec/m^3

5. SWEC Calculation Package RP-11700-131-1, Fuel Handling Accident - Based on 150, 200, 400, 500 and 800 Hours Delay, 1977

6. SWEC Calculation Package 12241 UR(B)-447-0, Combined Control Room Doses Due to BVPS #1 Fuel Handling Accident, Normal 500 cfm (unfiltered) Intake, 1987

7. SWEC Calculation Package 12241 UR(B)-456-0, 1, Combined BV1 - BV2 Control Room Habitability Due to Design Basis Accidents (except LOCA) at BV1, 1988

8. DLCo Calculation Package ERS-SFL-90-002-0, Safety Analysis of the Radiological Consequences of a Unit 1 Fuel Handling Accident With 60 Day Decay w/Filtration Delay, 1990

9. DLCo Calculation Package ERS-SFL-92-025-0, Safety Analysis of the Dose Consequences of a Fuel Handling Accident at BVPS 1 to the Common Control Room, 1992

10. HOLTEC Calculation (Performed to support the Unit 1 fuel pool re-rack effort, DCP-1516)

11. DLCo Calculation Package ERS-JTL-99-009-0, Safety Analysis of the Radiological Consequences of a Fuel Handling DBA at BVPS Unit 1, Control Room, EAB and LPZ Doses, 1999

12. BVPS Condition Report 972390 (Identified deficiency in dose calculation and provided for reviewing all DBA dose calculations, parameters, inputs and methodologies and revising where warranted), January 1998

13. BVPS LER-97-008-01, (Failure to Meet Single Active Failure criteria for Control Room Emergency Ventilation System Results in Entry Into T. S. 3.0.3), March 27, 1998

14. ORIGENS Computer Code, Radiation Safety Information Computational Center, U. S. Department of Energy, Oak Ridge TN (as used by Stone & Webster Engineering Corporation)

15. USEPA-520/1-88-020, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion (Federal Guidance Report No. 11), 1988

16. DOE/EH-0070, U. S. Department of Energy, External Dose-Rate Conversion Factors for Calculation of Dose to the Public, July 1988

17. BVPS Calculation Package ERS-JTL-99-009-1, Safety Analysis of the Radiological Consequences of a Fuel Handling DBA at BVPS Unit 1, Control Room, EAB and LPZ Doses, 2000

REFERENCES (continued)

18. BVPS Calculation Package ERS-SFL-96-004-2, TRAILS_PC: Transport of radioactive Material in Linear Systems, PC version v1.0b, 2000

19. USEPA-402-R-93-081, External Exposure to Radionuclides in Air, Water, and Soil (Federal Guidance Report No. 12), 1993

20. BVPS Unit 1 Technical Specification 3/4.9.3, Decay Time

21. DLCo NED Response to EM #116252, Design Basis Accident Parameter Verification - Unit 1, August 1, 1998

22. DLCo letter ND1NEM:1144, Letter from M. Siegel, Manager Nuclear Engineering Department to D. Piccione, Project Engineer, Stone & Webster Engineering Corporation, Data for use in performing the joint control room dose analysis, April 7, 1987

23. SWEC Calculation Package 12241 UR(B)-478-0, Design Reactor Core Inventory (3.96% Initial Enrichment) and Associated Equilibrium Primary and Secondary Coolant Activities for BVPS, 1999

24. SWEC Calculation Package 12241 UR(B)-479-0, Radiological Source Terms for Accident Analyses - Composite Equilibrium Reactor Core Inventory (3.6% - 5% Initial Enrichment) and the Associated Design Primary and Secondary Coolant Activities for BVPS, 1999

25. BVPS Unit 2 UFSAR Chapter 6, Table 6.4-1, Control Room Envelope Ventilation Design Parameters (Note: This UFSAR section is applicable to the Unit 1 {common} control room)

26. BVPS Operating Procedure 1/2OM-44A.4A.A, Post Control Room Habitability System Actuation/Recovery

27. BVPS Unit 1 Abnormal Operating Procedure 1OM-53C.4.1.49.1, Irradiated Fuel Damage

28. USNRC Safety Evaluation Report, Amendment No. 144 to Facility Operating License No. DPR-66, Docket No. 50-334 (and associated license amendment request 1A-162)

29. USNRC Safety Evaluation Report, Amendment No. 144 to Facility Operating License No. DPR-66, Docket No. 50-334 (and associated license amendment request 1A-162)

30. BVPS Health Physics Manual, Chapter 4 RIP 2.19, DRMS, Area Monitoring Subsystem

31. SWEC Calculation Package 12241 UR(B)-462-0, Digital Radiation Monitoring System Set Point Compilation, 1978

32. DLCo Letter, Beaver Valley Power Station Unit No. 2, Docket No. 50-412, License No. NPF-73, Offsite Dose Analysis to Support Operating License Change Request No. 57, February 25, 1992

33. BVPS Unit 2 License Amendment 57 and Accompanying USNRC Safety Evaluation, Docket No. 50-412, Issued September 28, 1993

34. BVPS Unit 1 UFSAR Chapter 2, Table 2.2-12, Main Control Room X/Q Values

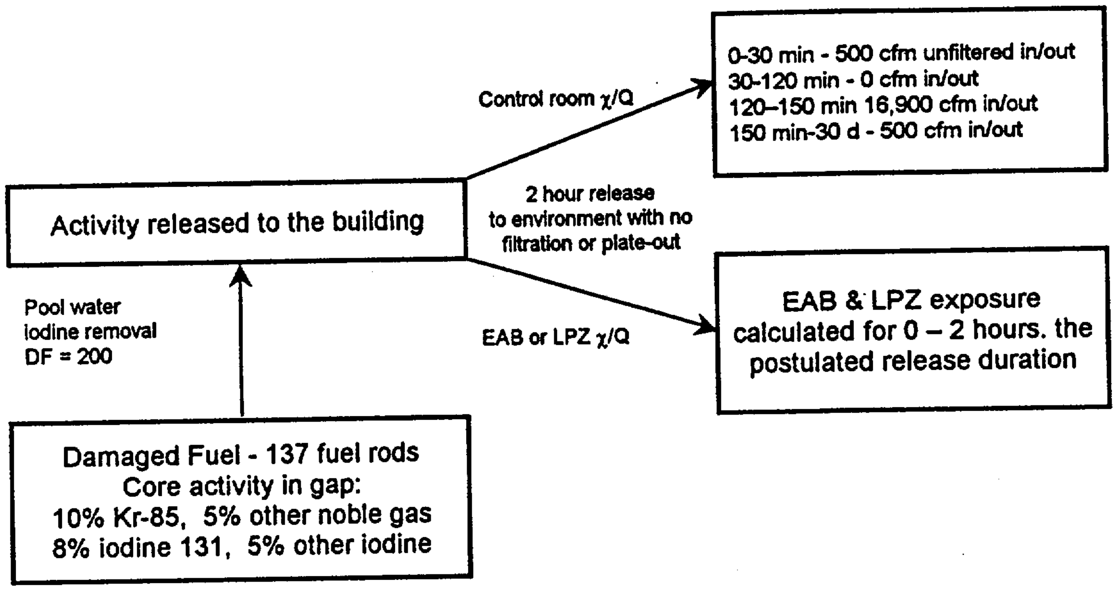
35. DLCo Calculation Package ERS-SFL-96-021-2, Regulatory Guide 1.145 Short-Term Accident X/Q Values for EAB and LPZ, Unit 1 and Unit 2, Based on 1986-1995 Observations, 1997

REFERENCES (continued)

36. BVPS Unit 1 UFSAR Chapter 2, Tables 2.2-11a, 0.5% Accident Analysis 0 to 2 hour X/Q Values at the Exclusion Area Boundary, and 2.2-11b, 0.5% Accident Analysis X/Q Values for Various Time Periods at the Low Population Zone Boundary

37. Title 10 Code of Federal Regulations, Part 50.67, Accident Source Terms

ATTACHMENT 1
Fuel Handling Accident Release Model



Fuel Handling Accident Release Source Term Determination

	A	B	C	D	E	F	G	H	I	J	K	L	M
1						Fuel Handling Accident Release Source Term Determination							
2													
3													=Col H * Col I * (J5/264
4				=Col B / Col C		=MAX(Col D:Col E)		=Col F * Col G		=Col H * Col I			(J5/264) * 1E+06
5–7		Core 100 hr decay (Ci)	Assemblies in core	(1) UR(B)-479 Core Avg.	(2) UR(B)-482 Max. Burn	Max. of Col D, C (Ci)	Radial Peaking	Max. Assembly (Ci)	Gap Fraction	Max. Assembly Gap Act (Ci)	Pool Release Fraction	No. Rods Damaged	137 Rods, Bldg Activity (μCi)
8	Kr-85m	3.64E+00	157	2.32E-02	1.93E-02	2.32E-02	1.65	3.83E-02	0.05	1.91E-03	1.00	137	9.93E+02
9	Kr-85	7.59E+05	157	4.83E+03	6.60E+03	6.60E+03	1.65	1.09E+04	0.10	1.09E+03	1.00	137	5.65E+08
10													
11	Xe-131m	9.19E+05	157	5.85E+03	6.41E+03	6.41E+03	1.65	1.06E+04	0.05	5.29E+02	1.00	137	2.74E+08
12	Xe-133m	1.92E+06	157	1.22E+04	1.22E+04	1.22E+04	1.65	2.02E+04	0.05	1.01E+03	1.00	137	5.24E+08
13	Xe-133	1.03E+08	157	6.56E+05	6.47E+05	6.56E+05	1.65	1.08E+06	0.05	5.41E+04	1.00	137	2.81E+10
14	Xe-135m	6.04E+02	157	3.85E+00	3.82E+00	3.85E+00	1.65	6.35E+00	0.05	3.17E-01	1.00	137	1.65E+05
15	Xe-135	1.99E+05	157	1.27E+03	1.25E+03	1.27E+03	1.65	2.09E+03	0.05	1.05E+02	1.00	137	5.43E+07
16													
17	I-131	5.23E+07	157	3.33E+05	3.38E+05	3.38E+05	1.65	5.58E+05	0.08	4.46E+04	0.005	137	1.16E+08
18	I-132	4.41E+07	157	2.81E+05	2.81E+05	2.81E+05	1.65	4.64E+05	0.05	2.32E+04	0.005	137	6.02E+07
19	I-133	5.45E+06	157	3.47E+04	3.42E+04	3.47E+04	1.65	5.73E+04	0.05	2.86E+03	0.005	137	7.43E+06
20	I-135	3.71E+03	157	2.36E+01	2.34E+01	2.36E+01	1.65	3.90E+01	0.05	1.95E+00	0.005	137	5.06E+03

21 Note: Nuclide activity is selected for either a maximum burn assembly or core average assembly, whichever is higher. The differences are small and this is conservative.

22 [1]Value from SWEC 12241/11700-UR(B)-479 - This calculation provides activities for the average burn assembly considering bounding uranium enrichment for each nuclide.

23 [2]Value from SWEC 12241/11700-UR(B)-482 - This calculation provides activities for the maximum burn assembly considering bounding uranium enrichment for each nuclide.

25 $(1-.999999) = e^{-\lambda t}$ where t = 7200s

26 $\lambda =$ 1.92E-03 s^{-1} release rate constant

28 Assumptions:

29 Per Regulatory Guide 1.183.

30 Iodine pool DF = 200

31 Radial peaking = 1.65 (This is conservative. AST guidance specifies that an actual value be used - 1.65 is from SG 25)

32 Gap fractions listed in Column I

TRAILS_PC Input File for Control Room, LPZ and EAB dose

```
'L1 ',11,1.0E-3,1.0E-4,1.0E-5,1.0E-4,1.0,1.0E-3,3.5E-4,2,24,1,2
'Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (U1_FHA.TXT)'
'C1 ','N/A ',0.0,0.0,0.0,0.0
'C2 ','FHB ',1.92E-3,0.0,0.0,0.0
'CR ',1.73E5,10.0,10.0,0.0,0.0,0.0
'PRD',24*0.0
'PRD',24*0.0
'INI',1.0,' ',24*0.0
'INI',1.0,'uCi ',0.0,9.93E2,5.65E8,4*0.0,
2.74E8,5.24E8,2.81E10,1.65E5,5.43E7,2*0.0
1.16E8,6.02E7,7.43E6,0.0,5.06E3,5*0.0
'INI',1.0,' ',24*0.0
'TIM',1200.,1800.,5400.,6600.,7200.,9000.,14400.,28800.,86400.,3.456E5,2.592E6
'XPR',11*0.0
'XPR',11*0.0
'XPR',2*50.0,2*0.0,0.0,1.690E3,5*50.0
'XRM',11*0.0
'XRM',5*1.0,6*0.0
'XRM',2*50.0,2*0.0,0.0,1.690E3,5*50.0
'XRF',11*0.0
'XRF',11*0.0
'XRF',11*0.0
'XOQEB',5*1.04,6*0.0
'XBREB',5*3.5,6*0.0
'XOQLZ',5*6.04,6*0.0
'XBRLZ',5*3.5,6*0.0
'XOQ',5*4.30,6*0.0
'XBR',11*1.0
'OCC',9*1.0,0.6,0.4
```

Modeling:
This models the release following a fuel handling accident.

1. Release is modeled as constant flow rate, exponential.
2. Activity in gap is 5% except Kr-85 - 10% and I-131 - 8%.
3. 100% of the noble gas gap activity from broken rods is released to the building.
4. 0.5% of the iodine gap activity from broken rods is released to the building.

Changes this revision:

1. Used guidance provided in NRC Regulatory Guide 1.183.
2. Remove any credit for pre-release iodine filtration.
3. Used bounding assumptions for control room habitability ventilation system parameters.
4. Credited the post-activity release control room habitability ventilation system purge feature.
5. Used the radioactivity released assuming from 137 damaged rods.

TRAILS12 -- Transport of Radioactive Material in Linear Systems, V1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT) ***PROGENY INGROWTH ON ***

	COMP: N/A	COMP: FHB	COMP: Control Room VOLUME: 1.730E+05 Cu.Ft.
INITIAL:	0.000E+00 Kr-85m	9.930E+02 Kr-85m uCi	0.000E+00 Kr-85m
	0.000E+00 Kr-85	5.650E+08 Kr-85	0.000E+00 Kr-85
	0.000E+00 Xe-131m	2.740E+08 Xe-131m	0.000E+00 Xe-131m
	0.000E+00 Xe-133m	5.240E+08 Xe-133m	0.000E+00 Xe-133m
	0.000E+00 Xe-133	2.810E+10 Xe-133	0.000E+00 Xe-133
	0.000E+00 Xe-135m	1.650E+05 Xe-135m	0.000E+00 Xe-135m
	0.000E+00 Xe-135	5.430E+07 Xe-135	0.000E+00 Xe-135
	0.000E+00 I-131	1.160E+08 I-131	0.000E+00 I-131
	0.000E+00 I-132	6.020E+07 I-132	0.000E+00 I-132
	0.000E+00 I-133	7.430E+06 I-133	0.000E+00 I-133
	0.000E+00 I-135	5.060E+03 I-135	0.000E+00 I-135
ACT MULT (to uCi):	1.000E+00	1.000E+00	1.000E+00

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT) ***PROGENY INGROWTH ON ***

REMOVAL:			0.000E+00 1/sec			1.920E-03 1/sec			1.000E+01 cfm	
NUC Grp 1 REL FR:				0.000E+00			0.000E+00		INTAKE REDUCT: 0.000E+00	
NUC Grp 2 REL FR:				0.000E+00			0.000E+00		INTAKE REDUCT: 0.000E+00	
NUC Grp 3 REL FR:				0.000E+00			0.000E+00		INTAKE REDUCT: 0.000E+00	
MULTIPLIERS====>										
STEP	TIME	XPR	XREM	XRF	XPR	XREM	XRF	XPR	XREM	XRF
1	1.200E+03	0.00	0.00	0.00	0.00	1.00	0.00	50.0	50.0	0.00
2	1.800E+03	0.00	0.00	0.00	0.00	1.00	0.00	50.0	50.0	0.00
3	5.400E+03	0.00	0.00	0.00	0.00	1.00	0.00	0.00	0.00	0.00
4	6.600E+03	0.00	0.00	0.00	0.00	1.00	0.00	0.00	0.00	0.00
5	7.200E+03	0.00	0.00	0.00	0.00	1.00	0.00	0.00	0.00	0.00
6	9.000E+03	0.00	0.00	0.00	0.00	0.00	0.00	1.690E+03	1.690E+03	0.00
7	1.440E+04	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
8	2.880E+04	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
9	8.640E+04	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
10	3.456E+05	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
11	2.592E+06	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00

		-------- CONTROL ROOM --------			--- EXCLUSION AREA BOUNDARY ---		--- LOW POPULATION ZONE ---	
		X/Q s/M3	Breathing M3/s	Occupancy	X/Q s/M3	Breathing M3/s	X/Q s/M3	Breathing M3/s
		1.000E-03	3.500E-04	1.000E+00	1.000E-03	1.000E-04	1.000E-05	1.000E-04
MULTIPLIERS====>								
STEP	TIME,s							
1	1.200E+03	4.30	1.00	1.00	1.04	3.50	6.04	3.50
2	1.800E+03	4.30	1.00	1.00	1.04	3.50	6.04	3.50
3	5.400E+03	4.30	1.00	1.00	1.04	3.50	6.04	3.50
4	6.600E+03	4.30	1.00	1.00	1.04	3.50	6.04	3.50
5	7.200E+03	4.30	1.00	1.00	1.04	3.50	6.04	3.50
6	9.000E+03	0.00	1.00	1.00	0.00	0.00	0.00	0.00
7	1.440E+04	0.00	1.00	1.00	0.00	0.00	0.00	0.00
8	2.880E+04	0.00	1.00	1.00	0.00	0.00	0.00	0.00
9	8.640E+04	0.00	1.00	1.00	0.00	0.00	0.00	0.00
10	3.456E+05	0.00	1.00	0.600	0.00	0.00	0.00	0.00
11	2.592E+06	0.00	1.00	0.400	0.00	0.00	0.00	0.00

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT) ***PROGENY INGROWTH ON ***

		N/A		FHB			AVERAGE	----------CONTROL ROOM----------		
		CURRENT	INTEGRD	CURRENT	INTEGRD	RELEASED	RELEASE	CURRENT	CURRENT	INTEGRD
STEP	TIME	uCi	uCi-sec	uCi	uCi-sec	uCi	uCi/sec	uCi	uCi/cc	uCi-sec
Kr-85m	INITIAL	0.000E+00		9.930E+02				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	9.418E+01	4.579E+05	8.791E+02	7.326E-01	8.450E-01	1.725E-10	5.162E+02
2	0.5000 h	0.000E+00	0.000E+00	2.900E+01	3.320E+04	6.375E+01	1.062E-01	8.630E-01	1.762E-10	5.124E+02
3	1.5000 h	0.000E+00	0.000E+00	2.474E-02	1.476E+04	2.834E+01	7.873E-03	7.393E-01	1.509E-10	2.878E+03
4	1.8333 h	0.000E+00	0.000E+00	2.346E-03	1.141E+01	2.190E-02	1.825E-05	7.021E-01	1.433E-10	8.646E+02
5	2.0000 h	0.000E+00	0.000E+00	7.226E-04	8.272E-01	1.588E-03	2.647E-06	6.842E-01	1.397E-10	4.159E+02
6	2.5000 h	0.000E+00	0.000E+00	6.688E-04	1.252E+00	0.000E+00	0.000E+00	3.380E-02	6.899E-12	3.892E+02
7	4.0000 h	0.000E+00	0.000E+00	5.303E-04	3.223E+00	0.000E+00	0.000E+00	2.066E-02	4.217E-12	1.441E+02
8	8.0000 h	0.000E+00	0.000E+00	2.856E-04	5.693E+00	0.000E+00	0.000E+00	5.560E-03	1.135E-12	1.656E+02
9	24.0000 h	0.000E+00	0.000E+00	2.402E-05	6.086E+00	0.000E+00	0.000E+00	2.917E-05	5.955E-15	6.068E+01
10	96.0000 h	0.000E+00	0.000E+00	3.488E-10	5.589E-01	0.000E+00	0.000E+00	1.602E-15	3.270E-25	3.201E-01
11	720.0000 h	0.000E+00	0.000E+00	0.000E+00	8.116E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.757E-11
Kr-85m	TOTALS		0.000E+00		5.059E+05	9.713E+02				5.948E+03
Kr-85	INITIAL	0.000E+00		5.650E+08				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	5.642E+07	2.649E+11	5.086E+08	4.238E+05	5.014E+05	1.024E-04	3.038E+08
2	0.5000 h	0.000E+00	0.000E+00	1.783E+07	2.010E+10	3.859E+07	6.432E+04	5.257E+05	1.073E-04	3.082E+08
3	1.5000 h	0.000E+00	0.000E+00	1.775E+04	9.277E+09	1.781E+07	4.948E+03	5.257E+05	1.073E-04	1.893E+09
4	1.8333 h	0.000E+00	0.000E+00	1.773E+03	8.323E+06	1.598E+04	1.332E+01	5.257E+05	1.073E-04	6.309E+08
5	2.0000 h	0.000E+00	0.000E+00	5.602E+02	6.316E+05	1.213E+03	2.021E+00	5.257E+05	1.073E-04	3.154E+08
6	2.5000 h	0.000E+00	0.000E+00	5.602E+02	1.008E+06	0.000E+00	0.000E+00	2.805E+04	5.727E-06	3.057E+08
7	4.0000 h	0.000E+00	0.000E+00	5.602E+02	3.025E+06	0.000E+00	0.000E+00	2.163E+04	4.415E-06	1.334E+08
8	8.0000 h	0.000E+00	0.000E+00	5.602E+02	8.067E+06	0.000E+00	0.000E+00	1.081E+04	2.206E-06	2.246E+08
9	24.0000 h	0.000E+00	0.000E+00	5.601E+02	3.227E+07	0.000E+00	0.000E+00	6.741E+02	1.376E-07	2.104E+08
10	96.0000 h	0.000E+00	0.000E+00	5.598E+02	1.451E+08	0.000E+00	0.000E+00	2.547E-03	5.200E-13	1.399E+07
11	720.0000 h	0.000E+00	0.000E+00	5.573E+02	1.255E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.288E+01
Kr-85	TOTALS		0.000E+00		2.957E+11	5.650E+08				4.339E+09
Xe-131m	INITIAL	0.000E+00		2.740E+08				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	2.734E+07	1.284E+11	2.466E+08	2.055E+05	2.430E+05	4.960E-05	1.472E+08
2	0.5000 h	0.000E+00	0.000E+00	8.636E+06	9.738E+09	1.870E+07	3.116E+04	2.547E+05	5.199E-05	1.493E+08
3	1.5000 h	0.000E+00	0.000E+00	8.578E+03	4.492E+09	8.624E+06	2.396E+03	2.541E+05	5.186E-05	9.157E+08
4	1.8333 h	0.000E+00	0.000E+00	8.559E+02	4.021E+06	7.720E+03	6.433E+00	2.539E+05	5.182E-05	3.048E+08
5	2.0000 h	0.000E+00	0.000E+00	2.704E+02	3.049E+05	5.854E+02	9.756E-01	2.538E+05	5.180E-05	1.523E+08
6	2.5000 h	0.000E+00	0.000E+00	2.700E+02	4.864E+05	0.000E+00	0.000E+00	1.352E+04	2.761E-06	1.475E+08
7	4.0000 h	0.000E+00	0.000E+00	2.691E+02	1.456E+06	0.000E+00	0.000E+00	1.039E+04	2.121E-06	6.420E+07
8	8.0000 h	0.000E+00	0.000E+00	2.665E+02	3.856E+06	0.000E+00	0.000E+00	5.142E+03	1.050E-06	1.074E+08
9	24.0000 h	0.000E+00	0.000E+00	2.563E+02	1.505E+07	0.000E+00	0.000E+00	3.085E+02	6.298E-08	9.896E+07
10	96.0000 h	0.000E+00	0.000E+00	2.152E+02	6.095E+07	0.000E+00	0.000E+00	9.793E-04	1.999E-13	6.316E+06
11	720.0000 h	0.000E+00	0.000E+00	4.717E+01	2.488E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.005E+01
Xe-131m	TOTALS		0.000E+00		1.430E+11	2.739E+08				2.094E+09

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT) ***PROGENY INGROWTH ON ***

STEP	TIME	N/A CURRENT uCi	N/A INTEGRD uCi-sec	FHB CURRENT uCi	FHB INTEGRD uCi-sec	RELEASED uCi	AVERAGE RELEASE uCi/sec	CONTROL ROOM CURRENT uCi	CONTROL ROOM CURRENT uCi/cc	CONTROL ROOM INTEGRD uCi-sec
Xe-133m	INITIAL	0.000E+00		5.240E+08				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	5.210E+07	2.453E+11	4.710E+08	3.925E+05	4.634E+05	9.459E-05	2.809E+08
2	0.5000 h	0.000E+00	0.000E+00	1.643E+07	1.854E+10	3.560E+07	5.934E+04	4.847E+05	9.895E-05	2.845E+08
3	1.5000 h	0.000E+00	0.000E+00	1.614E+04	8.531E+09	1.638E+07	4.550E+03	4.784E+05	9.766E-05	1.734E+09
4	1.8333 h	0.000E+00	0.000E+00	1.605E+03	7.557E+06	1.451E+04	1.209E+01	4.763E+05	9.723E-05	5.728E+08
5	2.0000 h	0.000E+00	0.000E+00	5.061E+02	5.712E+05	1.097E+03	1.828E+00	4.753E+05	9.701E-05	2.855E+08
6	2.5000 h	0.000E+00	0.000E+00	5.027E+02	9.079E+05	0.000E+00	0.000E+00	2.519E+04	5.143E-06	2.758E+08
7	4.0000 h	0.000E+00	0.000E+00	4.929E+02	2.688E+06	0.000E+00	0.000E+00	1.904E+04	3.887E-06	1.187E+08
8	8.0000 h	0.000E+00	0.000E+00	4.676E+02	6.914E+06	0.000E+00	0.000E+00	9.028E+03	1.843E-06	1.932E+08
9	24.0000 h	0.000E+00	0.000E+00	3.786E+02	2.428E+07	0.000E+00	0.000E+00	4.561E+02	9.309E-08	1.654E+08
10	96.0000 h	0.000E+00	0.000E+00	1.465E+02	6.337E+07	0.000E+00	0.000E+00	6.673E-04	1.362E-13	8.799E+06
11	720.0000 h	0.000E+00	0.000E+00	3.909E-02	3.999E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.287E+01
Xe-133m	TOTALS		0.000E+00		2.725E+11	5.230E+08				3.919E+09
Xe-133	INITIAL	0.000E+00		2.810E+10				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	2.801E+09	1.317E+13	2.528E+10	2.107E+07	2.490E+07	5.083E-03	1.509E+10
2	0.5000 h	0.000E+00	0.000E+00	8.843E+08	9.975E+11	1.915E+09	3.192E+06	2.608E+07	5.325E-03	1.530E+10
3	1.5000 h	0.000E+00	0.000E+00	8.758E+05	4.598E+11	8.828E+08	2.452E+05	2.594E+07	5.296E-03	9.365E+10
4	1.8333 h	0.000E+00	0.000E+00	8.730E+04	4.104E+08	7.879E+05	6.566E+02	2.590E+07	5.286E-03	3.110E+10
5	2.0000 h	0.000E+00	0.000E+00	2.756E+04	3.109E+07	5.969E+04	9.949E+01	2.587E+07	5.282E-03	1.553E+10
6	2.5000 h	0.000E+00	0.000E+00	2.749E+04	4.955E+07	0.000E+00	0.000E+00	1.377E+06	2.811E-04	1.503E+10
7	4.0000 h	0.000E+00	0.000E+00	2.727E+04	1.478E+08	0.000E+00	0.000E+00	1.053E+06	2.150E-04	6.522E+09
8	8.0000 h	0.000E+00	0.000E+00	2.668E+04	3.884E+08	0.000E+00	0.000E+00	5.150E+05	1.051E-04	1.083E+10
9	24.0000 h	0.000E+00	0.000E+00	2.447E+04	1.472E+09	0.000E+00	0.000E+00	2.946E+04	6.013E-06	9.775E+09
10	96.0000 h	0.000E+00	0.000E+00	1.654E+04	5.249E+09	0.000E+00	0.000E+00	7.528E-02	1.537E-11	5.930E+08
11	720.0000 h	0.000E+00	0.000E+00	5.357E+02	1.050E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.515E+03
Xe-133	TOTALS		0.000E+00		1.464E+13	2.808E+10				2.134E+11
Xe-135m	INITIAL	0.000E+00		1.650E+05				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	6.731E+03	5.934E+07	1.139E+05	9.494E+01	7.455E+01	1.522E-08	5.172E+04
2	0.5000 h	0.000E+00	0.000E+00	1.364E+03	2.017E+06	3.872E+03	6.453E+00	4.950E+01	1.011E-08	3.662E+04
3	1.5000 h	0.000E+00	0.000E+00	1.122E-01	5.139E+05	9.866E+02	2.741E-01	3.945E+00	8.053E-10	6.311E+04
4	1.8333 h	0.000E+00	0.000E+00	5.846E-03	4.271E+01	8.200E-02	6.833E-05	1.989E+00	4.060E-10	3.388E+03
5	2.0000 h	0.000E+00	0.000E+00	1.422E-03	1.871E+00	3.592E-03	5.987E-06	1.499E+00	3.059E-10	1.036E+03
6	2.5000 h	0.000E+00	0.000E+00	8.496E-04	1.938E+00	0.000E+00	0.000E+00	4.495E-02	9.176E-12	7.255E+02
7	4.0000 h	0.000E+00	0.000E+00	5.694E-04	3.555E+00	0.000E+00	0.000E+00	2.215E-02	4.521E-12	1.616E+02
8	8.0000 h	0.000E+00	0.000E+00	3.722E-04	6.663E+00	0.000E+00	0.000E+00	7.225E-03	1.475E-12	1.912E+02
9	24.0000 h	0.000E+00	0.000E+00	6.952E-05	1.039E+01	0.000E+00	0.000E+00	8.418E-05	1.718E-14	9.238E+01
10	96.0000 h	0.000E+00	0.000E+00	3.657E-08	2.385E+00	0.000E+00	0.000E+00	1.674E-13	3.417E-23	1.089E+00
11	720.0000 h	0.000E+00	0.000E+00	1.396E-36	1.255E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.166E-09
Xe-135m	TOTALS		0.000E+00		6.187E+07	1.188E+05				1.570E+05

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT) ***PROGENY INGROWTH ON ***

		N/A		FHB			AVERAGE	----------CONTROL ROOM----------		
		CURRENT	INTEGRD	CURRENT	INTEGRD	RELEASED	RELEASE	CURRENT	CURRENT	INTEGRD
STEP	TIME	uCi	uCi-sec	uCi	uCi-sec	uCi	uCi/sec	uCi	uCi/cc	uCi-sec
Xe-135	INITIAL	0.000E+00		5.430E+07				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	5.287E+06	2.525E+10	4.848E+07	4.040E+04	4.720E+04	9.636E-06	2.871E+07
2	0.5000 h	0.000E+00	0.000E+00	1.650E+06	1.874E+09	3.598E+06	5.996E+03	4.886E+04	9.973E-06	2.882E+07
3	1.5000 h	0.000E+00	0.000E+00	1.522E+03	8.490E+08	1.630E+06	4.528E+02	4.528E+04	9.243E-06	1.694E+08
4	1.8333 h	0.000E+00	0.000E+00	1.482E+02	7.079E+05	1.359E+03	1.133E+00	4.415E+04	9.012E-06	5.365E+07
5	2.0000 h	0.000E+00	0.000E+00	4.625E+01	5.253E+04	1.009E+02	1.681E-01	4.359E+04	8.898E-06	2.632E+07
6	2.5000 h	0.000E+00	0.000E+00	4.452E+01	8.168E+04	0.000E+00	0.000E+00	2.239E+03	4.571E-07	2.507E+07
7	4.0000 h	0.000E+00	0.000E+00	3.972E+01	2.272E+05	0.000E+00	0.000E+00	1.540E+03	3.144E-07	1.009E+07
8	8.0000 h	0.000E+00	0.000E+00	2.930E+01	4.931E+05	0.000E+00	0.000E+00	5.678E+02	1.159E-07	1.403E+07
9	24.0000 h	0.000E+00	0.000E+00	8.673E+00	9.759E+05	0.000E+00	0.000E+00	1.048E+01	2.140E-09	8.041E+06
10	96.0000 h	0.000E+00	0.000E+00	3.623E-02	4.086E+05	0.000E+00	0.000E+00	1.656E-07	3.380E-17	1.513E+05
11	720.0000 h	0.000E+00	0.000E+00	8.705E-23	1.714E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.389E-03
Xe-135	TOTALS		0.000E+00		2.798E+10	5.371E+07				3.643E+08
I-131	INITIAL	0.000E+00		1.160E+08				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	1.157E+07	5.436E+10	1.044E+08	8.698E+04	1.028E+05	2.099E-05	6.231E+07
2	0.5000 h	0.000E+00	0.000E+00	3.654E+06	4.121E+09	7.912E+06	1.319E+04	1.078E+05	2.200E-05	6.319E+07
3	1.5000 h	0.000E+00	0.000E+00	3.625E+03	1.900E+09	3.648E+06	1.013E+03	1.074E+05	2.192E-05	3.873E+08
4	1.8333 h	0.000E+00	0.000E+00	3.616E+02	1.699E+06	3.262E+03	2.718E+00	1.073E+05	2.189E-05	1.288E+08
5	2.0000 h	0.000E+00	0.000E+00	1.142E+02	1.288E+05	2.473E+02	4.121E-01	1.072E+05	2.188E-05	6.433E+07
6	2.5000 h	0.000E+00	0.000E+00	1.140E+02	2.054E+05	0.000E+00	0.000E+00	5.710E+03	1.166E-06	6.229E+07
7	4.0000 h	0.000E+00	0.000E+00	1.134E+02	6.139E+05	0.000E+00	0.000E+00	4.378E+03	8.937E-07	2.708E+07
8	8.0000 h	0.000E+00	0.000E+00	1.118E+02	1.621E+06	0.000E+00	0.000E+00	2.157E+03	4.403E-07	4.518E+07
9	24.0000 h	0.000E+00	0.000E+00	1.055E+02	6.256E+06	0.000E+00	0.000E+00	1.270E+02	2.593E-08	4.128E+07
10	96.0000 h	0.000E+00	0.000E+00	8.147E+01	2.410E+07	0.000E+00	0.000E+00	3.708E-04	7.569E-14	2.583E+06
11	720.0000 h	0.000E+00	0.000E+00	8.660E+00	7.297E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.542E+00
I-131	TOTALS		0.000E+00		6.049E+10	1.159E+08				8.843E+08
I-132	INITIAL	0.000E+00		6.020E+07				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	5.437E+06	2.733E+10	5.248E+07	4.373E+04	4.925E+04	1.005E-05	3.033E+07
2	0.5000 h	0.000E+00	0.000E+00	1.634E+06	1.898E+09	3.644E+06	6.074E+03	4.906E+04	1.001E-05	2.949E+07
3	1.5000 h	0.000E+00	0.000E+00	1.204E+03	8.149E+08	1.565E+06	4.346E+02	3.629E+04	7.408E-06	1.525E+08
4	1.8333 h	0.000E+00	0.000E+00	1.087E+02	5.465E+05	1.049E+03	8.743E-01	3.282E+04	6.700E-06	4.143E+07
5	2.0000 h	0.000E+00	0.000E+00	3.267E+01	3.795E+04	7.286E+01	1.214E-01	3.121E+04	6.372E-06	1.921E+07
6	2.5000 h	0.000E+00	0.000E+00	2.810E+01	5.459E+04	0.000E+00	0.000E+00	1.433E+03	2.925E-07	1.740E+07
7	4.0000 h	0.000E+00	0.000E+00	1.788E+01	1.221E+05	0.000E+00	0.000E+00	7.029E+02	1.435E-07	5.534E+06
8	8.0000 h	0.000E+00	0.000E+00	5.356E+00	1.496E+05	0.000E+00	0.000E+00	1.052E+02	2.148E-08	4.532E+06
9	24.0000 h	0.000E+00	0.000E+00	4.313E-02	6.347E+04	0.000E+00	0.000E+00	5.285E-02	1.079E-11	7.974E+05
10	96.0000 h	0.000E+00	0.000E+00	1.626E-11	5.152E+02	0.000E+00	0.000E+00	7.535E-17	1.538E-26	4.007E+02
11	720.0000 h	0.000E+00	0.000E+00	0.000E+00	1.943E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.714E-13
I-132	TOTALS		0.000E+00		3.004E+10	5.768E+07				3.012E+08

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT) ***PROGENY INGROWTH ON ***

		N/A		FHB			AVERAGE	----------CONTROL ROOM----------		
		CURRENT	INTEGRD	CURRENT	INTEGRD	RELEASED	RELEASE	CURRENT	CURRENT	INTEGRD
STEP	TIME	uCi	uCi-sec	uCi	uCi-sec	uCi	uCi/sec	uCi	uCi/cc	uCi-sec
I-133	INITIAL	0.000E+00		7.430E+06				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	7.338E+05	3.471E+09	6.664E+06	5.553E+03	6.534E+03	1.334E-06	3.966E+06
2	0.5000 h	0.000E+00	0.000E+00	2.306E+05	2.608E+08	5.008E+05	8.346E+02	6.812E+03	1.391E-06	4.005E+06
3	1.5000 h	0.000E+00	0.000E+00	2.221E+02	1.194E+08	2.293E+05	6.368E+01	6.589E+03	1.345E-06	2.412E+07
4	1.8333 h	0.000E+00	0.000E+00	2.193E+01	1.037E+05	1.992E+02	1.660E-01	6.516E+03	1.330E-06	7.863E+06
5	2.0000 h	0.000E+00	0.000E+00	6.892E+00	7.796E+03	1.497E+01	2.495E-02	6.480E+03	1.323E-06	3.899E+06
6	2.5000 h	0.000E+00	0.000E+00	6.778E+00	1.230E+04	0.000E+00	0.000E+00	3.401E+02	6.942E-08	3.750E+06
7	4.0000 h	0.000E+00	0.000E+00	6.448E+00	3.570E+04	0.000E+00	0.000E+00	2.494E+02	5.091E-08	1.579E+06
8	8.0000 h	0.000E+00	0.000E+00	5.643E+00	8.693E+04	0.000E+00	0.000E+00	1.091E+02	2.227E-08	2.444E+06
9	24.0000 h	0.000E+00	0.000E+00	3.311E+00	2.519E+05	0.000E+00	0.000E+00	3.993E+00	8.150E-10	1.830E+06
10	96.0000 h	0.000E+00	0.000E+00	3.006E-01	3.252E+05	0.000E+00	0.000E+00	1.370E-06	2.797E-16	6.952E+04
11	720.0000 h	0.000E+00	0.000E+00	2.799E-10	3.247E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.386E-02
I-133	TOTALS		0.000E+00		3.852E+09	7.394E+06				5.353E+07
I-135	INITIAL	0.000E+00		5.060E+03				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	4.879E+02	2.346E+06	4.504E+03	3.753E+00	4.364E+00	8.909E-10	2.659E+03
2	0.5000 h	0.000E+00	0.000E+00	1.515E+02	1.726E+05	3.314E+02	5.523E-01	4.495E+00	9.176E-10	2.658E+03
3	1.5000 h	0.000E+00	0.000E+00	1.359E-01	7.767E+04	1.491E+02	4.142E-02	4.048E+00	8.262E-10	1.536E+04
4	1.8333 h	0.000E+00	0.000E+00	1.310E-02	6.298E+01	1.209E-01	1.008E-04	3.909E+00	7.979E-10	4.773E+03
5	2.0000 h	0.000E+00	0.000E+00	4.068E-03	4.634E+00	8.897E-03	1.483E-05	3.841E+00	7.840E-10	2.325E+03
6	2.5000 h	0.000E+00	0.000E+00	3.860E-03	7.134E+00	0.000E+00	0.000E+00	1.945E-01	3.970E-11	2.200E+03
7	4.0000 h	0.000E+00	0.000E+00	3.298E-03	1.929E+01	0.000E+00	0.000E+00	1.281E-01	2.615E-11	8.586E+02
8	8.0000 h	0.000E+00	0.000E+00	2.168E-03	3.879E+01	0.000E+00	0.000E+00	4.209E-02	8.592E-12	1.113E+03
9	24.0000 h	0.000E+00	0.000E+00	4.050E-04	6.054E+01	0.000E+00	0.000E+00	4.904E-04	1.001E-13	5.382E+02
10	96.0000 h	0.000E+00	0.000E+00	2.130E-07	1.390E+01	0.000E+00	0.000E+00	9.753E-13	1.991E-22	6.344E+00
11	720.0000 h	0.000E+00	0.000E+00	8.132E-36	7.313E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.262E-08
I-135	TOTALS		0.000E+00		2.596E+06	4.984E+03				3.249E+04

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT)   ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	EDE DOSE RATE mrem/h	CEDE DOSE mrem	TEDE DOSE mrem
Kr-85m										
0.3333 h	2.53E-05	0.00E+00	2.53E-05	1.47E-06	0.00E+00	1.47E-06	1.47E-07	8.64E-07	0.00E+00	1.47E-07
0.5000 h	1.84E-06	0.00E+00	1.84E-06	1.07E-07	0.00E+00	1.07E-07	1.46E-07	8.83E-07	0.00E+00	1.46E-07
1.5000 h	8.16E-07	0.00E+00	8.16E-07	4.74E-08	0.00E+00	4.74E-08	8.18E-07	7.56E-07	0.00E+00	8.18E-07
1.8333 h	6.31E-10	0.00E+00	6.31E-10	3.66E-11	0.00E+00	3.66E-11	2.46E-07	7.18E-07	0.00E+00	2.46E-07
2.0000 h	4.57E-11	0.00E+00	4.57E-11	2.66E-12	0.00E+00	2.66E-12	1.18E-07	7.00E-07	0.00E+00	1.18E-07
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.11E-07	3.46E-08	0.00E+00	1.11E-07
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.09E-08	2.11E-08	0.00E+00	4.09E-08
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.71E-08	5.69E-09	0.00E+00	4.71E-08
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.72E-08	2.98E-11	0.00E+00	1.72E-08
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.46E-11	1.64E-21	0.00E+00	5.46E-11
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.00E-21	0.00E+00	0.00E+00	2.00E-21
TOTALS	2.80E-05	0.00E+00	2.80E-05	1.62E-06	0.00E+00	1.62E-06	1.69E-06		0.00E+00	1.69E-06
Kr-85										
0.3333 h	2.34E-01	0.00E+00	2.34E-01	1.36E-02	0.00E+00	1.36E-02	1.38E-03	8.18E-03	0.00E+00	1.38E-03
0.5000 h	1.77E-02	0.00E+00	1.77E-02	1.03E-03	0.00E+00	1.03E-03	1.40E-03	8.57E-03	0.00E+00	1.40E-03
1.5000 h	8.18E-03	0.00E+00	8.18E-03	4.75E-04	0.00E+00	4.75E-04	8.57E-03	8.57E-03	0.00E+00	8.57E-03
1.8333 h	7.34E-06	0.00E+00	7.34E-06	4.26E-07	0.00E+00	4.26E-07	2.86E-03	8.57E-03	0.00E+00	2.86E-03
2.0000 h	5.57E-07	0.00E+00	5.57E-07	3.24E-08	0.00E+00	3.24E-08	1.43E-03	8.57E-03	0.00E+00	1.43E-03
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.38E-03	4.58E-04	0.00E+00	1.38E-03
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.04E-04	3.53E-04	0.00E+00	6.04E-04
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.02E-03	1.76E-04	0.00E+00	1.02E-03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.53E-04	1.10E-05	0.00E+00	9.53E-04
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.80E-05	4.15E-11	0.00E+00	3.80E-05
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.58E-11	0.00E+00	0.00E+00	9.58E-11
TOTALS	2.60E-01	0.00E+00	2.60E-01	1.51E-02	0.00E+00	1.51E-02	1.96E-02		0.00E+00	1.96E-02
Xe-131m										
0.3333 h	3.70E-01	0.00E+00	3.70E-01	2.15E-02	0.00E+00	2.15E-02	2.18E-03	1.29E-02	0.00E+00	2.18E-03
0.5000 h	2.80E-02	0.00E+00	2.80E-02	1.63E-03	0.00E+00	1.63E-03	2.21E-03	1.36E-02	0.00E+00	2.21E-03
1.5000 h	1.29E-02	0.00E+00	1.29E-02	7.51E-04	0.00E+00	7.51E-04	1.35E-02	1.35E-02	0.00E+00	1.35E-02
1.8333 h	1.16E-05	0.00E+00	1.16E-05	6.72E-07	0.00E+00	6.72E-07	4.51E-03	1.35E-02	0.00E+00	4.51E-03
2.0000 h	8.78E-07	0.00E+00	8.78E-07	5.10E-08	0.00E+00	5.10E-08	2.25E-03	1.35E-02	0.00E+00	2.25E-03
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.18E-03	7.20E-04	0.00E+00	2.18E-03
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.49E-04	5.53E-04	0.00E+00	9.49E-04
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.59E-03	2.74E-04	0.00E+00	1.59E-03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.46E-03	1.64E-05	0.00E+00	1.46E-03
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.60E-05	5.21E-11	0.00E+00	5.60E-05
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.19E-10	0.00E+00	0.00E+00	1.19E-10
TOTALS	4.11E-01	0.00E+00	4.11E-01	2.39E-02	0.00E+00	2.39E-02	3.09E-02		0.00E+00	3.09E-02

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT)   ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	EDE DOSE RATE mrem/h	CEDE DOSE mrem	TEDE DOSE mrem
Xe-133m										
0.3333 h	2.49E+00	0.00E+00	2.49E+00	1.45E-01	0.00E+00	1.45E-01	1.46E-02	8.70E-02	0.00E+00	1.46E-02
0.5000 h	1.88E-01	0.00E+00	1.88E-01	1.09E-02	0.00E+00	1.09E-02	1.48E-02	9.10E-02	0.00E+00	1.48E-02
1.5000 h	8.66E-02	0.00E+00	8.66E-02	5.03E-03	0.00E+00	5.03E-03	9.04E-02	8.98E-02	0.00E+00	9.04E-02
1.8333 h	7.67E-05	0.00E+00	7.67E-05	4.46E-06	0.00E+00	4.46E-06	2.99E-02	8.94E-02	0.00E+00	2.99E-02
2.0000 h	5.80E-06	0.00E+00	5.80E-06	3.37E-07	0.00E+00	3.37E-07	1.49E-02	8.92E-02	0.00E+00	1.49E-02
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.44E-02	4.73E-03	0.00E+00	1.44E-02
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.19E-03	3.57E-03	0.00E+00	6.19E-03
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.01E-02	1.69E-03	0.00E+00	1.01E-02
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.62E-03	8.56E-05	0.00E+00	8.62E-03
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.75E-04	1.25E-10	0.00E+00	2.75E-04
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.69E-10	0.00E+00	0.00E+00	2.69E-10
TOTALS	2.77E+00	0.00E+00	2.77E+00	1.61E-01	0.00E+00	1.61E-01	2.04E-01		0.00E+00	2.04E-01
Xe-133										
0.3333 h	1.52E+02	0.00E+00	1.52E+02	8.82E+00	0.00E+00	8.82E+00	8.94E-01	5.31E+00	0.00E+00	8.94E-01
0.5000 h	1.15E+01	0.00E+00	1.15E+01	6.68E-01	0.00E+00	6.68E-01	9.07E-01	5.57E+00	0.00E+00	9.07E-01
1.5000 h	5.30E+00	0.00E+00	5.30E+00	3.08E-01	0.00E+00	3.08E-01	5.55E+00	5.54E+00	0.00E+00	5.55E+00
1.8333 h	4.73E-03	0.00E+00	4.73E-03	2.75E-04	0.00E+00	2.75E-04	1.84E+00	5.53E+00	0.00E+00	1.84E+00
2.0000 h	3.59E-04	0.00E+00	3.59E-04	2.08E-05	0.00E+00	2.08E-05	9.21E-01	5.52E+00	0.00E+00	9.21E-01
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.91E-01	2.94E-01	0.00E+00	8.91E-01
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.87E-01	2.25E-01	0.00E+00	3.87E-01
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.42E-01	1.10E-01	0.00E+00	6.42E-01
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.79E-01	6.29E-03	0.00E+00	5.79E-01
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.11E-02	1.61E-08	0.00E+00	2.11E-02
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.59E-08	0.00E+00	0.00E+00	3.59E-08
TOTALS	1.69E+02	0.00E+00	1.69E+02	9.80E+00	0.00E+00	9.80E+00	1.26E+01		0.00E+00	1.26E+01
Xe-135m										
0.3333 h	8.95E-03	0.00E+00	8.95E-03	5.20E-04	0.00E+00	5.20E-04	4.01E-05	2.08E-04	0.00E+00	4.01E-05
0.5000 h	3.04E-04	0.00E+00	3.04E-04	1.77E-05	0.00E+00	1.77E-05	2.84E-05	1.38E-04	0.00E+00	2.84E-05
1.5000 h	7.75E-05	0.00E+00	7.75E-05	4.50E-06	0.00E+00	4.50E-06	4.89E-05	1.10E-05	0.00E+00	4.89E-05
1.8333 h	6.44E-09	0.00E+00	6.44E-09	3.74E-10	0.00E+00	3.74E-10	2.63E-06	5.55E-06	0.00E+00	2.63E-06
2.0000 h	2.82E-10	0.00E+00	2.82E-10	1.64E-11	0.00E+00	1.64E-11	8.03E-07	4.18E-06	0.00E+00	8.03E-07
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.62E-07	1.25E-07	0.00E+00	5.62E-07
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.25E-07	6.18E-08	0.00E+00	1.25E-07
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.48E-07	2.02E-08	0.00E+00	1.48E-07
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.16E-08	2.35E-10	0.00E+00	7.16E-08
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.06E-10	4.67E-19	0.00E+00	5.06E-10
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.71E-19	0.00E+00	0.00E+00	6.71E-19
TOTALS	9.33E-03	0.00E+00	9.33E-03	5.42E-04	0.00E+00	5.42E-04	1.22E-04		0.00E+00	1.22E-04

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT) ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	EDE DOSE RATE mrem/h	CEDE DOSE mrem	TEDE DOSE mrem
Xe-135										
0.3333 h	2.23E+00	0.00E+00	2.23E+00	1.29E-01	0.00E+00	1.29E-01	1.30E-02	7.70E-02	0.00E+00	1.30E-02
0.5000 h	1.65E-01	0.00E+00	1.65E-01	9.60E-03	0.00E+00	9.60E-03	1.31E-02	7.97E-02	0.00E+00	1.31E-02
1.5000 h	7.49E-02	0.00E+00	7.49E-02	4.35E-03	0.00E+00	4.35E-03	7.67E-02	7.39E-02	0.00E+00	7.67E-02
1.8333 h	6.24E-05	0.00E+00	6.24E-05	3.63E-06	0.00E+00	3.63E-06	2.43E-02	7.20E-02	0.00E+00	2.43E-02
2.0000 h	4.63E-06	0.00E+00	4.63E-06	2.69E-07	0.00E+00	2.69E-07	1.19E-02	7.11E-02	0.00E+00	1.19E-02
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.14E-02	3.65E-03	0.00E+00	1.14E-02
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.57E-03	2.51E-03	0.00E+00	4.57E-03
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.36E-03	9.26E-04	0.00E+00	6.36E-03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.64E-03	1.71E-05	0.00E+00	3.64E-03
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.11E-05	2.70E-13	0.00E+00	4.11E-05
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.33E-13	0.00E+00	0.00E+00	4.33E-13
TOTALS	2.47E+00	0.00E+00	2.47E+00	1.43E-01	0.00E+00	1.43E-01	1.65E-01		0.00E+00	1.65E-01
I-131										
0.3333 h	7.33E+00	1.25E+03	1.26E+03	4.26E-01	7.26E+01	7.30E+01	4.31E-02	2.56E-01	1.46E+02	1.47E+02
0.5000 h	5.55E-01	9.47E+01	9.53E+01	3.23E-02	5.50E+00	5.53E+00	4.38E-02	2.69E-01	1.49E+02	1.49E+02
1.5000 h	2.56E-01	4.37E+01	4.39E+01	1.49E-02	2.54E+00	2.55E+00	2.68E-01	2.68E-01	9.10E+02	9.11E+02
1.8333 h	2.29E-04	3.91E-02	3.93E-02	1.33E-05	2.27E-03	2.28E-03	8.92E-02	2.67E-01	3.03E+02	3.03E+02
2.0000 h	1.74E-05	2.96E-03	2.98E-03	1.01E-06	1.72E-04	1.73E-04	4.45E-02	2.67E-01	1.51E+02	1.51E+02
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.31E-02	1.42E-02	1.46E+02	1.46E+02
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.87E-02	1.09E-02	6.36E+01	6.37E+01
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.13E-02	5.38E-03	1.06E+02	1.06E+02
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.86E-02	3.17E-04	9.70E+01	9.71E+01
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.07E-03	9.24E-10	3.64E+00	3.64E+00
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.09E-09	0.00E+00	7.09E-06	7.09E-06
TOTALS	8.14E+00	1.39E+03	1.40E+03	4.73E-01	8.06E+01	8.11E+01	6.12E-01		2.08E+03	2.08E+03
I-132										
0.3333 h	2.26E+01	7.28E+00	2.99E+01	1.31E+00	4.23E-01	1.73E+00	1.29E-01	7.53E-01	8.26E-01	9.54E-01
0.5000 h	1.57E+00	5.05E-01	2.07E+00	9.11E-02	2.94E-02	1.20E-01	1.25E-01	7.50E-01	8.03E-01	9.28E-01
1.5000 h	6.73E-01	2.17E-01	8.90E-01	3.91E-02	1.26E-02	5.17E-02	6.47E-01	5.55E-01	4.15E+00	4.80E+00
1.8333 h	4.52E-04	1.46E-04	5.97E-04	2.62E-05	8.45E-06	3.47E-05	1.76E-01	5.02E-01	1.13E+00	1.30E+00
2.0000 h	3.14E-05	1.01E-05	4.15E-05	1.82E-06	5.87E-07	2.41E-06	8.15E-02	4.77E-01	5.23E-01	6.04E-01
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.39E-02	2.19E-02	4.74E-01	5.47E-01
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.35E-02	1.07E-02	1.51E-01	1.74E-01
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.92E-02	1.61E-03	1.23E-01	1.43E-01
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.39E-03	8.08E-07	2.17E-02	2.51E-02
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.02E-06	1.15E-21	6.54E-06	7.57E-06
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.70E-22	0.00E+00	6.22E-21	7.19E-21
TOTALS	2.48E+01	8.00E+00	3.28E+01	1.44E+00	4.65E-01	1.91E+00	1.28E+00		8.20E+00	9.48E+00

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 1 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (1FHA_AST.TXT) ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	EDE DOSE RATE mrem/h	CEDE DOSE mrem	TEDE DOSE mrem
I-133										
0.3333 h	7.55E-01	1.42E+01	1.49E+01	4.38E-02	8.24E-01	8.68E-01	4.43E-03	2.63E-02	1.66E+00	1.66E+00
0.5000 h	5.67E-02	1.07E+00	1.12E+00	3.29E-03	6.19E-02	6.52E-02	4.47E-03	2.74E-02	1.67E+00	1.68E+00
1.5000 h	2.60E-02	4.88E-01	5.14E-01	1.51E-03	2.84E-02	2.99E-02	2.69E-02	2.65E-02	1.01E+01	1.01E+01
1.8333 h	2.26E-05	4.24E-04	4.47E-04	1.31E-06	2.46E-05	2.59E-05	8.78E-03	2.62E-02	3.29E+00	3.30E+00
2.0000 h	1.70E-06	3.19E-05	3.36E-05	9.84E-08	1.85E-06	1.95E-06	4.36E-03	2.61E-02	1.63E+00	1.63E+00
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.19E-03	1.37E-03	1.57E+00	1.57E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.76E-03	1.00E-03	6.60E-01	6.62E-01
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.73E-03	4.39E-04	1.02E+00	1.02E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.04E-03	1.61E-05	7.65E-01	7.67E-01
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.66E-05	5.51E-12	1.74E-02	1.75E-02
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.07E-11	0.00E+00	3.99E-09	4.00E-09
TOTALS	8.37E-01	1.57E+01	1.66E+01	4.86E-02	9.14E-01	9.63E-01	5.98E-02		2.24E+01	2.24E+01
I-135										
0.3333 h	1.38E-03	2.02E-03	3.40E-03	8.01E-05	1.17E-04	1.97E-04	8.03E-06	4.75E-05	2.34E-04	2.42E-04
0.5000 h	1.01E-04	1.48E-04	2.50E-04	5.89E-06	8.62E-06	1.45E-05	8.03E-06	4.89E-05	2.34E-04	2.42E-04
1.5000 h	4.57E-05	6.68E-05	1.12E-04	2.65E-06	3.88E-06	6.53E-06	4.64E-05	4.40E-05	1.35E-03	1.40E-03
1.8333 h	3.70E-08	5.41E-08	9.12E-08	2.15E-09	3.14E-09	5.29E-09	1.44E-05	4.25E-05	4.19E-04	4.34E-04
2.0000 h	2.72E-09	3.98E-09	6.71E-09	1.58E-10	2.31E-10	3.90E-10	7.02E-06	4.18E-05	2.04E-04	2.11E-04
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.65E-06	2.11E-06	1.93E-04	2.00E-04
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.59E-06	1.39E-06	7.55E-05	7.80E-05
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.36E-06	4.58E-07	9.78E-05	1.01E-04
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.63E-06	5.33E-09	4.73E-05	4.89E-05
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.15E-08	1.06E-17	3.35E-07	3.46E-07
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.52E-17	0.00E+00	4.43E-16	4.59E-16
TOTALS	1.53E-03	2.23E-03	3.76E-03	8.87E-05	1.30E-04	2.18E-04	9.81E-05		2.86E-03	2.95E-03
ALL NUCLIDES										
0.3333 h	1.88E+02	1.27E+03	1.46E+03	1.09E+01	7.38E+01	8.48E+01	1.10E+00	6.53E+00	1.49E+02	1.50E+02
0.5000 h	1.41E+01	9.63E+01	1.10E+02	8.18E-01	5.59E+00	6.41E+00	1.11E+00	6.80E+00	1.51E+02	1.52E+02
1.5000 h	6.44E+00	4.44E+01	5.08E+01	3.74E-01	2.58E+00	2.95E+00	6.68E+00	6.57E+00	9.25E+02	9.31E+02
1.8333 h	5.60E-03	3.96E-02	4.52E-02	3.25E-04	2.30E-03	2.63E-03	2.18E+00	6.50E+00	3.07E+02	3.09E+02
2.0000 h	4.21E-04	3.00E-03	3.42E-03	2.45E-05	1.74E-04	1.99E-04	1.08E+00	6.47E+00	1.53E+02	1.54E+02
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.04E+00	3.41E-01	1.48E+02	1.49E+02
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.43E-01	2.54E-01	6.45E+01	6.49E+01
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.14E-01	1.20E-01	1.07E+02	1.08E+02
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.28E-01	6.75E-03	9.78E+01	9.85E+01
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.26E-02	1.72E-08	3.66E+00	3.68E+00
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.85E-08	0.00E+00	7.09E-06	7.13E-06
TOTALS	2.08E+02	1.41E+03	1.62E+03	1.21E+01	8.20E+01	9.41E+01	1.50E+01		2.11E+03	2.12E+03

ATTACHMENT C-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 73

Safety Analysis of the Radiological Consequences of a Fuel Handling DBA at BVPS Unit 2, Control Room, EAB and LPZ Doses

RTL: A9.621A
Form: RE 1.103-1

Beaver Valley Power Station

Health Physics Section

	REVISION [redacted] 3	
Subject **Safety Analysis of the Radiological Consequences of a Fuel Handling DBA at BVPS Unit 2, Control Room, EAB and LPZ Doses**	**ERS-SFL-89-019**	PAGE 1 OF **24**

Reference
HPM RP/RIP______ EPP______ T/S______ EM______ DCP______

Review Category

☒ RSC Required ☐ RSC Not Required ☐ 10 CFR 50.59 Required

Unit 1 ☐ Unit 2 ☒

Purpose
This calculation package documents an analysis of the postulated dose in the common control room, at the Exclusion Area Boundary (EAB), and at the Low Population Zone (LPZ) following a Fuel Handling DBA at BVPS Unit 2. This revision follows the guidance provided in USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

NOTE: This calculation package documents the evaluation described above. This package does not, alone, provide for any revision in a structure, system or component; nor changes in procedures, tests, or experiments described in the plant licensing basis. The data and/or conclusions of this package shall not be extended to other procedures without explicit concurrence from Health Physics.

Rev.	Signatures	Description
	by date *chk* date *app* date	
3	*by* John T. Lebda 10-18-00 date *chk* 10/18/00 date *app* FOR THE RSC (12-00) 10/27/00 date	Used maximum number of damaged fuel assembly rods (137) that could result, assuming the worst-case conditions that may be encountered in either the fuel building or the reactor containment building at Unit 2. Uses the current analysis assumptions and input parameters to the extent that they are consistent with those specified in Regulatory Guide 1.183 - changed pool water Iodine DF, activity fraction in-gap, submersion dose conversion factors and calculated TEDE.
2	*by* date *chk* date *app* date	See previous revisions for associated signature pages. This is a revised cover sheet.
1	*by* date *chk* date *app* date	See previous revisions for associated signature pages. This is a revised cover sheet.
0	*by* date *chk* date *app* date	**Checklist**: ☒ Purpose ☒ Assumptions ☒ Methodology ☒ Input Data ☒ Results ☒ References **Attachments**: ☒ Data Sheets ☒ Illustrations ☒ Printouts ☐ Code Listings

☒ BV RECORDS CENTER	☐ Supv, Rad Ops-1	☒ Author: J. Lebda
☒ CALCULATION FILE	☐ Supv, Rad Ops-2	☒ NS&L, A. Dometrovich
☒ MGR, Health Physics	☐ Supv, Effl & Rad Waste	☒ Mark Duranko
☒ Supv, Rad Eng & Health	☐ Training Section	☐ ______

DISCUSSION

General

This calculation determines control room operator, exclusion area boundary (EAB) and low population zone (LPZ) radiation doses following a fuel handling design basis accident (FHA DBA) at Beaver Valley Unit 2. The plant parameters and assumptions used herein are consistent with the plant design basis as revised in accordance with Regulatory Guide 1.183[1], and they provide results that define the upper bound of the accident dose consequences. As a design basis calculation, this is not intended to show what the expected doses would be, but rather what they might be if the plant is operated at "worst case" design limits (operating at the NRC License limits) and using "worst case" accident conditions. The combination of circumstances that would result in the associated maximum dose is unlikely to occur.

This revision of the FHA was requested by the Licensing Section. It is performed to demonstrate that the design case accident dose will remain within the regulatory dose limits and criteria with certain changes made to plant operation and configuration. Additionally, this analysis represents Beaver Valley Unit 2's first use of the alternative source term as provided in the recently issued Regulatory Guide 1.183[1]. As such, the design basis of the facility is changed, and NRC review is required prior to implementation of any of the configuration/operational changes supported by the analysis. The facility changes made herein are specific to the FHA scenario, and affect the activity available for release and pre-release treatment for this accident only. The characteristics of plant configuration and/or operation associated with the other design basis accidents are not changed. Thus the validity of each of the other design basis accidents as described in the facility UFSAR is unaffected and a clear, consistent and logical design basis is maintained. Refer to the Input Data and Assumptions section of this calculation package for additional information concerning the analysis methodology and assumption changes.

Fuel Handling Accident

This DBA is described in NRC Regulatory Guide 1.183[1] and NUREG-0800 Chapter 15, Section 15.0.1[2]. The accident occurs while moving a fuel assembly in either the fuel building fuel storage pool or in the reactor building containment cavity or transfer canal. The assembly is dropped, resulting in rupture of 137 fuel rods and release of radioactive iodine and noble gas into the pool water. The extent of damage has been determined by performing an analysis[3] using the limiting drop conditions and considering the weight of the dropped heavy load or the weight of the 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 assemblies has been considered.

As the iodine percolates to the pool surface much of it will be "scrubbed out" and remain in the water. This reduces the amount of iodine that is released from the damaged rods by a factor of 200 before it becomes airborne within the building. The noble gas activity released from the damaged rods is all released to the building with no removal by pool water scrubbing. After becoming airborne, the radioactivity is released to the environment assuming a constant air flow rate (exponential activity removal rate). The analysis model uses a conservatively calculated release rate constant that results in 99.9999% of the activity being released to the environment in the two hours immediately following the accident. Because the accident conditions may include having any of the reactor building containment penetrations open (including the equipment hatch or personnel airlock), and the release may be via any one or a combination of penetrations, the most restrictive release point atmospheric dispersion factor(s) are conservatively applied to the entire release. For this analysis to remain valid, the radioactivity release must be via one of these three points. Additionally, this analysis conservatively does not take credit for any pre-release filtration or iodine plate-out.

Calculation History – BVPS Unit 2

SWEC 12241 UR(B)-189-0 (1982)[4]
Initial calculation of environmental dose (EAB and LPZ). Superseded by SWEC 12241 UR(B)-189-1. This (and subsequent) calculation considers only an accident in the fuel building.

SWEC 12241 UR(B)-280-0 (1983)[5]
Initial calculation of Control Room operator dose. Superseded by SWEC 12241 UR(B)-295-0.

SWEC 12241 UR(B)-189-1 (1984)[6]
Revision to update atmospheric dispersion factor for the EAB. Superseded by ERS-SFL-89-019.

SWEC 12241 UR(B)-295-0 (1984)[7]
Calculated control room operator dose, with different assumptions for operator action to actuate the control room emergency ventilation system.

SWEC 12241 UR(B)-445-0 (1987)[8]
Calculated control room operator dose for the combined (Unit 1 + Unit 2) control room. Superseded by ERS-SFL-89-019-0.

ERS-SFL-89-007-0 (1989)[9]
Special calculation to assess effect of delaying control room isolation for 10 minutes following an FHA.

ERS-SFL-89-019-0 (1989)[10]
Calculated control room operator and environmental (EAB and LPZ) dose assuming an increased iodine 131 percentage (12%) of core activity in the rod gap.
***** Calculation of record for environmental dose *****

ERS-SFL-93-004-0 (1993)[11]
Calculated control room operator dose assuming that the normal control room ventilation system configuration is maintained for the accident duration (no credit for system filtration). Used updated atmospheric dispersion factors (chi/Q).
***** Calculation of record for control room operator dose *****

ERS-SFL-89-019-1 (1999)[12]
Used revised reactor core radionuclide inventory calculated using updated fuel parameters (high burn-up) and the computer code ORIGENS[13]; also, the core inventory used is determined by selecting the maximum activity from a range of core enrichments for each radioisotope. Used current, approved values for EAB and LPZ atmospheric dispersion factors (chi/Q); models release from the fuel building as a puff release; used updated ICRP 26/30 based dose quantities and dose conversion factors[14,15]. This calculation is intended to replace ERS-SFL-89-019-0[10] and ERS-SFL-93-004[11]; however, it has not been reviewed and accepted by the NRC. Consequently it is not a part of the Unit 2 design basis.

ERS-SFL-89-019-2 (2000)[16]
Assumed maximum of 617 fuel rods breached. Made slight adjustment to the dropped assembly radioactivity content by using the highest individual nuclide activity selected from between a core averaged assembly and a maximum burn assembly. All previous analyses used a core average assembly. This change is in addition to using the maximum individual nuclide activity selected over a range of core enrichment, first used in the previous revision of this calculation.

THIS CALCULATION - ERS-SFL-89-019-3 (2000)
Uses the maximum number of damaged fuel assembly rods (137) that could result, assuming the worst-case conditions that may be encountered in either the fuel building or the reactor containment building at Unit 2. The validity of this radiological consequence analysis is contingent upon NRC review and acceptance of the engineering analysis[3] that determined the maximum number of rods damaged. Uses the current analysis assumptions and input parameters to the extent that they are consistent with those specified in Regulatory Guide 1.183[1]. Changes made to conform with this Guide affect pool water iodine DF, activity fraction in-gap, reference source for submersion dose conversion factors and the dose quantity that is calculated. All changes are detailed in the Input Data and Assumptions section of this analysis.

METHODOLOGY

Overall Methodology

The methodology used in this analysis is similar to that used in previous analyses. This revision uses version 1.0b of the TRAILS_PC (for Transport of Radioactive mAterial In Linear Systems), PC version documented in Reference 17. This version is similar to the VAX-based version used in the current calculation of record for control room dose, and has been verified to produce like results. This newer version has the capability to model progeny in-growth, a feature used in this analysis. Also, the submersion dose conversion factors from DOE/EH-0700[14] (used in previous revisions of the FHA analysis) have been replaced with those from Federal Guidance Report No. 12[18]. This "b" version of TRAILS_PC has been designated as TRAILS12 to readily identify this change to the user. More details are provided with the TRAILS_PC input file in Attachment 3.

Fuel Handling Accident Modeling

The FHA model is relatively simple and is depicted in Attachment 1.

INPUT DATA AND ASSUMPTIONS

1.0 Assumptions

1.1 This analysis of the Unit 2 FHA is based on guidance provided in NUREG-0800, Chapter 15.0.1[2] and USNRC Regulatory Guide 1.183[1].

1.2 Minimum 100 hours between reactor shutdown (subcriticality) to the accident release. [Assumption]

This value is decreased from the current Technical Specification[19] value of 150 hours with the intent of supporting a change to this requirement.

1.3 Radioactivity release to the building occurs instantaneously, followed by a release to the environment for a two hour period. [1]

To model a constant air flow, a release rate constant which results in 99.9999% of the total radioactivity released in two hours is used. This approach is conservative particularly for the control room operator and EAB 0-2 hour doses. Release rate is important where control room radiation monitor response is needed, and the alarm time will be calculated. This calculation takes no credit for this protective feature.

Regulatory Guide 1.183[1] specifies that the EAB and LPZ doses be calculated for the two hour period that results in the highest dose. Because, per the Guidance, effectively all of the activity from the fuel handling accident Is released to the environment in the first two hours, only this time period need be considered.

1.4 The radioactivity released to the building is then released to the environment with no credit for activity removal by filtration or plate-out. [Assumption]

While some of the release pathways may be filtered, this will not be credited in this analysis. Assuming no filtration is a conservative assumption.

2.0. Input Data

2.1 Core activity in gap: [1]

I-131	=	8%
Other iodines	=	5%
Kr-85	=	10%
Other noble gases	=	5%

2.2 Radial peaking factor = 1.65 [References 4–12,16]

For events that do not involve the entire core, Regulatory Guide 1.183[1] provides "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". 1.65 is the value used in previous analyses and is slightly higher than the facility-specific value. Continued used of this value is slightly conservative.

2.3 Number of assemblies in core = 157 [20]

2.4 Number of ruptured fuel rods = 137 [3]

2.5 Number of rods in assembly = 264 [20]

2.6 Fuel pool iodine DF = 200 [1]

2.7 Core Inventory at T = 100 hours, and release to the building: [21, 22]

Calculated in Attachment 2

Regulatory Guide 1.183[1] Appendix A provides "Radionuclides that should be considered include xenons, kryptons, halogens, cesiums and rubidiums". Additionally, Section 4.1.1 provides in part, for the EAB and beyond, "The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity".

Although this statement is not specifically repeated for the control room, Section 4.2.2 provides "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". Considering this guidance, the source term from the previous revision of this calculation (includes kryptons, xenons and iodines) is deemed appropriate for continued use based on the following.

Cesium and rubidium are particulates, therefore the activity released from the fuel will be retained in the pool water. However, cesium may be produced in the building and the environment as progeny of the I-135 – Xe-135 – Cs-135 decay chain. Because of the 100 hour pre-accident decay period, the 135 decay chain is the only one that has precursor isotopes of sufficiently long half-life to contribute to post-release cesium production. Cs-135 has a very long half-life and it's precursors have short half-lives. Consequently, the total activity of Cs-135 that could be produced from the released I-135 and Xe-135 is minute, and is not considered to be significant with regard to either submersion or internal dose. Post-release rubidium production will not contribute to dose. Kr-85 and Kr-85m are the only precursor isotopes that remain after the 100 hour pre-accident decay period, and these decay to stable Rb-85.

The halogen, Br-82, will be present in the fuel clad gap in small quantities even after the 100 hour pre-release decay period. The dose contribution from this isotope was evaluated and, assuming that pool scrubbing will occur similar to that for iodine, this contribution is not significant, i.e., will not change the dose values calculated in this analysis.

2.8 Pool release iodine composition = *N/A for this analysis* [N/A]

Regulatory Guide 1.183[1] provides specific fractions for the iodine release chemical form. These become important when determining removal by filtration, an assumption NOT made in this analysis.

2.9 Two hour duration release rate constant (λ) = 1.92E-03 s^{-1} [Assumption]

For a two hour release duration, 99.9999% of the radioactivity is assumed to be released in the two hours.

$$0.000001 = e^{-\lambda t}$$
$$\ln(0.000001) = \lambda t$$
$$13.82/t = \lambda$$
$$\text{where } t = 7200 \text{ seconds}, \lambda = 1.92\text{E-03 } s^{-1}$$

2.10 Control room air intake and exhaust flow rate = 500 cfm [23]

The control room ventilation system is assumed to remain in the normal mode operation for the entire duration of the analysis period (T = 0 – 30 days).

2.11 Control room volume = 1.73E+05 ft^3 [23]

2.12 Control room occupancy factors: [1]

0 to 24 hours = 1.0
24 hours to 4 days = 0.6
4 to 30 days = 0.4

2.13 Control room operator breathing rate = 3.5E-04 m^3/s [1]

2.14. Offsite breathing rates (0 to 8 hours) = 3.5E-04 m^3/s [1]

2.15 Atmospheric dispersion factors: [24]

Unit 2:

	EAB (s/m^3)	LPZ (s/m^3)	Control room (s/m^3)
0 – 2 hours	1.25E-03		
0 – 8 hours		6.04E-05	1.04E-03 Primary Auxiliary Building

Note: Because this release is conservatively modeled as a having a two hour duration, atmospheric dispersion factors beyond this time are unnecessary.

RESULTS AND CONCLUSIONS

The assumptions and data given above were used to build the TRAILS_PC (v1.0b) input file for the control room, LPZ and the EAB dose calculation. The input and output files are given in Attachment 3. The control room operator, EAB and LPZ calculated doses are summarized below.

	TEDE (rem)
Control room	1.4E+00
EAB	2.0E+00
LPZ	9.5E-02

Offsite doses are within the applicable regulatory limit of 10 CFR 50.67[25] of 25 rem TEDE at the EAB (for any 2-hour period) and LPZ (for the entire period of radioactive cloud passage), and are less than the more restrictive guidance criteria in the NUREG-0800 Section 15.0.1[2] and Regulatory Guide 1.183[1] of 6.3 rem TEDE for the 2 hour release duration. Control room operator doses (for the duration of the accident) are less than the 10 CFR 50.67[25] limit of 5 rem TEDE.

REFERENCES

1. USNRC Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, 2000
2. USNRC Standard Review Plan (NUREG-0800) SRP 15.0.1, Radiological Consequence Analyses Using Alternative Source Terms, 2000
3. HOLTEC Report No. HI-992343, Project No. 90844, Evaluation of Spent Fuel Assembly Drop Accidents in the Beaver Valley Power Station Reactor Core, 2000
4. SWEC Calculation Package 12241 UR(B)-189-0, FSAR Section 15.7.4 - Fuel Handling Accident: Releases and Doses, 1982
5. SWEC Calculation Package 12241 UR(B)-280-0, Control Room Dose Due to A Fuel Handling Accident - Normal Operational and Accident Ventilation Systems, 1983
6. SWEC Calculation Package 12241 UR(B)-189-1, FSAR Section 15.7.4 - Fuel Handling Accident: Releases and Doses, 1984
7. SWEC Calculation Package 12241 UR(B)-295, Control Room Doses Due to A Fuel Handling Accident as A Function of Operator Response Time, 1984
8. SWEC Calculation Package 12241 UR(B)-445-0, Control Room Habitability Due to Design Basis Accidents (Except LOCA) at BV2, 1987
9. DLCo Calculation Package ERS-SFL-89-007-0, Combined Control Room Doses Due to DBAs at Unit 2 With Delayed Isolation, 1989
10. DLCo Calculation Package ERS-SFL-89-019-0, Unit 2 Fuel Handling Accident Doses at EAB, LPZ, and Common Control Room, 1989
11. DLCo Calculation Package ERS-SFL-93-004-0, Safety Analysis of Consequences of Control Room Damper Response Delay (Limitorque 10 CFR 21) -- Unit 2 Accidents
12. DLCo Calculation Package ERS-SFL-89-019-1, Unit 2 Fuel Handling Accident Doses at EAB, LPZ, and Common Control Room, 1999
13. ORIGENS Computer Code, Radiation Safety Information Computational Center, U. S. Department of Energy, Oak Ridge TN (as used by Stone & Webster Engineering Corporation)
14. USEPA-520/1-88-020, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion (Federal Guidance Report No. 11), 1988
15. DOE/EH-0070, U. S. Department of Energy, External Dose-Rate Conversion Factors for Calculation of Dose to the Public, July 1988
16. DLCo Calculation Package ERS-SFL-89-019-2, Unit 2 Fuel Handling Accident Doses at EAB, LPZ, and Common Control Room, 2000
17. BVPS Calculation Package ERS-SFL-96-004-2, TRAILS_PC: Transport of Radioactive Material in Linear Systems, PC version v1.0b, 2000

REFERENCES (continued)

18. USEPA-402-R-93-081, External Exposure to Radionuclides in Air, Water, and Soil (Federal Guidance Report No. 12), 1993

19. BVPS Unit 2 Technical Specification 3/4.9.3, Decay Time

20. DLCo NED Response to EM #116965, Input Values for Core Inventory Calculation – Unit 2, October 6, 1998

21. SWEC Calculation Package 12241 UR(B)-478-0, Design Reactor Core Inventory (3.96% Initial Enrichment) and Associated Equilibrium Primary and Secondary Coolant Activities for BVPS, 1999

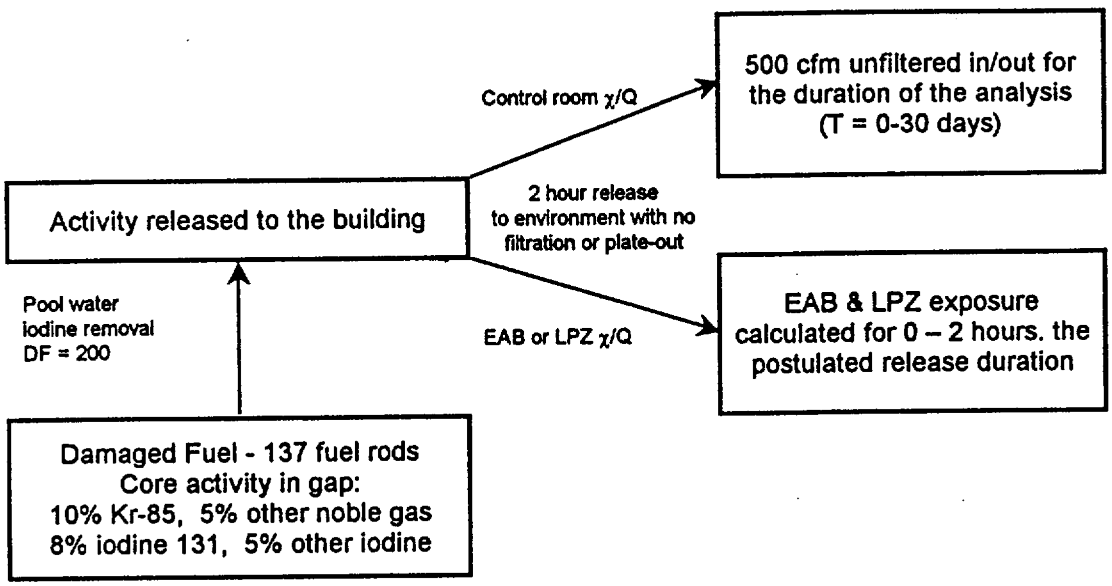
22. SWEC Calculation Package 12241 UR(B)-479-0, Radiological Source Terms for Accident Analyses - Composite Equilibrium Reactor Core Inventory (3.6% - 5% Initial Enrichment) and the Associated Design Primary and Secondary Coolant Activities for BVPS, 1999

23. BVPS Unit 2 UFSAR Chapter 6, Table 6.4-1, Control Room Envelope Ventilation Design Parameters (Note: This UFSAR section is applicable to the Unit 1 {common} control room)

24. BVPS Unit 2 UFSAR Chapter 15, Table 15.0-14, Accident Meteorological Parameters

25. Title 10 Code of Federal Regulations, Part 50.67,

ATTACHMENT 1
Fuel Handling Accident Release Model



	A	B	C	D	E	F	G	H	I	J	K	L	M
1					Fuel Handling Accident Release Source Term Determination								
2													
3													=Col H * Col I * (J5/264
4				=Col B / Col C		=MAX(Col D:Col E)		=Col F * Col G		=Col H * Col I			(J5/264) * 1E+06
5		Core		(1)	(2)	Max. of				Max. Assembly	Pool		137
6		100 hr decay	Assemblies	UR(B)-479	UR(B)-482	Col D, C	Radial	Max. Assembly	Gap	Gap Act	Release	No. Rods	Rods, Bldg Activity
7		(Ci)	in core	Core Avg.	Max. Burn	(Ci)	Peaking	(Ci)	Fraction	(Ci)	Fraction	Damaged	(μCi)
8	Kr-85m	3.64E+00	157	2.32E-02	1.93E-02	2.32E-02	1.65	3.83E-02	0.05	1.91E-03	1.00	137	9.93E+02
9	Kr-85	7.59E+05	157	4.83E+03	6.60E+03	6.60E+03	1.65	1.09E+04	0.10	1.09E+03	1.00	137	5.65E+08
10													
11	Xe-131m	9.19E+05	157	5.85E+03	6.41E+03	6.41E+03	1.65	1.06E+04	0.05	5.29E+02	1.00	137	2.74E+08
12	Xe-133m	1.92E+06	157	1.22E+04	1.22E+04	1.22E+04	1.65	2.02E+04	0.05	1.01E+03	1.00	137	5.24E+08
13	Xe-133	1.03E+08	157	6.56E+05	6.47E+05	6.56E+05	1.65	1.08E+06	0.05	5.41E+04	1.00	137	2.81E+10
14	Xe-135m	6.04E+02	157	3.85E+00	3.82E+00	3.85E+00	1.65	6.35E+00	0.05	3.17E-01	1.00	137	1.65E+05
15	Xe-135	1.99E+05	157	1.27E+03	1.25E+03	1.27E+03	1.65	2.09E+03	0.05	1.05E+02	1.00	137	5.43E+07
16													
17	I-131	5.23E+07	157	3.33E+05	3.38E+05	3.38E+05	1.65	5.58E+05	0.08	4.46E+04	0.005	137	1.16E+08
18	I-132	4.41E+07	157	2.81E+05	2.81E+05	2.81E+05	1.65	4.64E+05	0.05	2.32E+04	0.005	137	6.02E+07
19	I-133	5.45E+06	157	3.47E+04	3.42E+04	3.47E+04	1.65	5.73E+04	0.05	2.86E+03	0.005	137	7.43E+06
20	I-135	3.71E+03	157	2.36E+01	2.34E+01	2.36E+01	1.65	3.90E+01	0.05	1.95E+00	0.005	137	5.06E+03

21 Note: Nuclide activity is selected for either a maximum burn assembly or core average assembly, whichever is higher. The differences are small and this is conservative.

22 [1]Value from SWEC 12241/11700-UR(B)-479 - This calculation provides activities for the average burn assembly considering bounding uranium enrichment for each nuclide.

23 [2]Value from SWEC 12241/11700-UR(B)-482 - This calculation provides activities for the maximum burn assembly considering bounding uranium enrichment for each nuclide.

24

25 $(1-.999999) = e^{-\lambda t}$ where t = 7200s

26 $\lambda =$ 1.92E-03 s^{-1} release rate constant

27

28 Assumptions:

29 Per Regulatory Guide 1.183.

30 Iodine pool DF = 200

31 Radial peaking = 1.65 (This is conservative. AST guidance specifies that an actual value be used - 1.65 is from SG 25)

32 Gap fractions listed in Column I

TRAILS_PC Input File for Control Room, LPZ (0-30 day) and EAB (0-2 hour) dose

```
'L1 ',11,1.0E-3,1.0E-4,1.0E-5,1.0E-4,1.0,1.0E-3,3.5E-4,2,24,1,2
'Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT)'
'C1 ','N/A ',0.0,0.0,0.0,0.0
'C2 ','FHB ',1.92E-3,0.0,0.0,0.0
'CR ',1.73E5,10.0,10.0,0.0,0.0,0.0
'PRD',24*0.0
'PRD',24*0.0
'INI',1.0,' ',24*0.0
'INI',1.0,'uCi ',0.0,9.93E2,5.65E8,4*0.0,
2.74E8,5.24E8,2.81E10,1.65E5,5.43E7,2*0.0
1.16E8,6.02E7,7.43E6,0.0,5.06E3,5*0.0
'INI',1.0,' ',24*0.0
'TIM',1200.,1800.,5400.,6600.,7200.,9000.,14400.,28800.,86400.,3.456E5,2.592E6
'XPR',11*0.0
'XPR',11*0.0
'XPR',11*50.0
'XRM',11*0.0
'XRM',5*1.0,6*0.0
'XRM',11*50.0
'XRF',11*0.0
'XRF',11*0.0
'XRF',11*0.0
'XOQEB',5*1.25,6*0.0
'XBREB',5*3.5,6*0.0
'XOQLZ',5*6.04,6*0.0
'XBRLZ',5*3.5,6*0.0
'XOQ',5*1.04,6*0.0
'XBR',11*1.0
'OCC',9*1.0,0.6,0.4
```

Modeling:
This models the release following a fuel handling accident.

1. Release is modeled as constant flow rate, exponential.
2. Activity in gap is 5% except Kr-85 - 10% and I-131 - 8%.
3. 100% of the noble gas gap activity from broken rods is released to the building.
4. 0.5% of the iodine gap activity from broken rods is released to the building.

Changes this revision:

1. Used guidance provided in NRC Regulatory Guide 1.183.
2. Remove any credit for pre-release iodine filtration.
3. Used the radioactivity released assuming from 137 damaged rods.

TRAILS12 -- Transport of Radioactive Material in Linear Systems, V1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

	COMP: N/A	COMP: FHB	COMP: Control Room VOLUME: 1.730E+05 Cu.Ft.
INITIAL:	0.000E+00 Kr-85m	9.930E+02 Kr-85m uCi	0.000E+00 Kr-85m
	0.000E+00 Kr-85	5.650E+08 Kr-85	0.000E+00 Kr-85
	0.000E+00 Xe-131m	2.740E+08 Xe-131m	0.000E+00 Xe-131m
	0.000E+00 Xe-133m	5.240E+08 Xe-133m	0.000E+00 Xe-133m
	0.000E+00 Xe-133	2.810E+10 Xe-133	0.000E+00 Xe-133
	0.000E+00 Xe-135m	1.650E+05 Xe-135m	0.000E+00 Xe-135m
	0.000E+00 Xe-135	5.430E+07 Xe-135	0.000E+00 Xe-135
	0.000E+00 I-131	1.160E+08 I-131	0.000E+00 I-131
	0.000E+00 I-132	6.020E+07 I-132	0.000E+00 I-132
	0.000E+00 I-133	7.430E+06 I-133	0.000E+00 I-133
	0.000E+00 I-135	5.060E+03 I-135	0.000E+00 I-135
ACT MULT (to uCi):	1.000E+00	1.000E+00	1.000E+00

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

REMOVAL:	0.000E+00 1/sec	1.920E-03 1/sec	1.000E+01 cfm
NUC Grp 1 REL FR:	0.000E+00	0.000E+00	INTAKE REDUCT: 0.000E+00
NUC Grp 2 REL FR:	0.000E+00	0.000E+00	INTAKE REDUCT: 0.000E+00
NUC Grp 3 REL FR:	0.000E+00	0.000E+00	INTAKE REDUCT: 0.000E+00

MULTIPLIERS====>

STEP	TIME	XPR	XREM	XRF	XPR	XREM	XRF	XPR	XREM	XRF
1	1.200E+03	0.00	0.00	0.00	0.00	1.00	0.00	50.0	50.0	0.00
2	1.800E+03	0.00	0.00	0.00	0.00	1.00	0.00	50.0	50.0	0.00
3	5.400E+03	0.00	0.00	0.00	0.00	1.00	0.00	50.0	50.0	0.00
4	6.600E+03	0.00	0.00	0.00	0.00	1.00	0.00	50.0	50.0	0.00
5	7.200E+03	0.00	0.00	0.00	0.00	1.00	0.00	50.0	50.0	0.00
6	9.000E+03	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
7	1.440E+04	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
8	2.880E+04	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
9	8.640E+04	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
10	3.456E+05	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
11	2.592E+06	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00

		-------- CONTROL ROOM --------			--- EXCLUSION AREA BOUNDARY ---		--- LOW POPULATION ZONE ---	
		X/Q s/M3	Breathing M3/s	Occupancy	X/Q s/M3	Breathing M3/s	X/Q s/M3	Breathing M3/s
		1.000E-03	3.500E-04	1.000E+00	1.000E-03	1.000E-04	1.000E-05	1.000E-04
MULTIPLIERS====>								
STEP	TIME,s							
1	1.200E+03	1.04	1.00	1.00	1.25	3.50	6.04	3.50
2	1.800E+03	1.04	1.00	1.00	1.25	3.50	6.04	3.50
3	5.400E+03	1.04	1.00	1.00	1.25	3.50	6.04	3.50
4	6.600E+03	1.04	1.00	1.00	1.25	3.50	6.04	3.50
5	7.200E+03	1.04	1.00	1.00	1.25	3.50	6.04	3.50
6	9.000E+03	0.00	1.00	1.00	0.00	0.00	0.00	0.00
7	1.440E+04	0.00	1.00	1.00	0.00	0.00	0.00	0.00
8	2.880E+04	0.00	1.00	1.00	0.00	0.00	0.00	0.00
9	8.640E+04	0.00	1.00	1.00	0.00	0.00	0.00	0.00
10	3.456E+05	0.00	1.00	0.600	0.00	0.00	0.00	0.00
11	2.592E+06	0.00	1.00	0.400	0.00	0.00	0.00	0.00

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

STEP	TIME	N/A CURRENT uCi	N/A INTEGRD uCi-sec	FHB CURRENT uCi	FHB INTEGRD uCi-sec	RELEASED uCi	AVERAGE RELEASE uCi/sec	CONTROL ROOM CURRENT uCi	CONTROL ROOM CURRENT uCi/cc	CONTROL ROOM INTEGRD uCi-sec
Kr-85m	INITIAL	0.000E+00		9.930E+02				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	9.418E+01	4.579E+05	8.791E+02	7.326E-01	2.044E-01	4.172E-11	1.249E+02
2	0.5000 h	0.000E+00	0.000E+00	2.900E+01	3.320E+04	6.375E+01	1.062E-01	2.087E-01	4.261E-11	1.239E+02
3	1.5000 h	0.000E+00	0.000E+00	2.474E-02	1.476E+04	2.834E+01	7.873E-03	1.563E-01	3.190E-11	6.518E+02
4	1.8333 h	0.000E+00	0.000E+00	2.346E-03	1.141E+01	2.190E-02	1.825E-05	1.401E-01	2.859E-11	1.776E+02
5	2.0000 h	0.000E+00	0.000E+00	7.226E-04	8.272E-01	1.588E-03	2.647E-06	1.326E-01	2.707E-11	8.179E+01
6	2.5000 h	0.000E+00	0.000E+00	6.688E-04	1.252E+00	0.000E+00	0.000E+00	1.126E-01	2.298E-11	2.202E+02
7	4.0000 h	0.000E+00	0.000E+00	5.303E-04	3.223E+00	0.000E+00	0.000E+00	6.880E-02	1.405E-11	4.800E+02
8	8.0000 h	0.000E+00	0.000E+00	2.856E-04	5.693E+00	0.000E+00	0.000E+00	1.852E-02	3.780E-12	5.517E+02
9	24.0000 h	0.000E+00	0.000E+00	2.402E-05	6.086E+00	0.000E+00	0.000E+00	9.716E-05	1.983E-14	2.021E+02
10	96.0000 h	0.000E+00	0.000E+00	3.488E-10	5.589E-01	0.000E+00	0.000E+00	5.335E-15	1.089E-24	1.066E+00
11	720.0000 h	0.000E+00	0.000E+00	0.000E+00	8.116E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.853E-11
Kr-85m	TOTALS		0.000E+00		5.059E+05	9.713E+02				2.615E+03
Kr-85	INITIAL	0.000E+00		5.650E+08				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	5.642E+07	2.649E+11	5.086E+08	4.238E+05	1.213E+05	2.476E-05	7.347E+07
2	0.5000 h	0.000E+00	0.000E+00	1.783E+07	2.010E+10	3.859E+07	6.432E+04	1.272E+05	2.596E-05	7.454E+07
3	1.5000 h	0.000E+00	0.000E+00	1.775E+04	9.277E+09	1.781E+07	4.948E+03	1.109E+05	2.264E-05	4.277E+08
4	1.8333 h	0.000E+00	0.000E+00	1.773E+03	8.323E+06	1.598E+04	1.332E+01	1.047E+05	2.137E-05	1.293E+08
5	2.0000 h	0.000E+00	0.000E+00	5.602E+02	6.316E+05	1.213E+03	2.021E+00	1.017E+05	2.076E-05	6.192E+07
6	2.5000 h	0.000E+00	0.000E+00	5.602E+02	1.008E+06	0.000E+00	0.000E+00	9.327E+04	1.904E-05	1.754E+08
7	4.0000 h	0.000E+00	0.000E+00	5.602E+02	3.025E+06	0.000E+00	0.000E+00	7.190E+04	1.468E-05	4.435E+08
8	8.0000 h	0.000E+00	0.000E+00	5.602E+02	8.067E+06	0.000E+00	0.000E+00	3.593E+04	7.335E-06	7.467E+08
9	24.0000 h	0.000E+00	0.000E+00	5.601E+02	3.227E+07	0.000E+00	0.000E+00	2.241E+03	4.575E-07	6.994E+08
10	96.0000 h	0.000E+00	0.000E+00	5.598E+02	1.451E+08	0.000E+00	0.000E+00	8.469E-03	1.729E-12	4.652E+07
11	720.0000 h	0.000E+00	0.000E+00	5.573E+02	1.255E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.758E+02
Kr-85	TOTALS		0.000E+00		2.957E+11	5.650E+08				2.878E+09
Xe-131m	INITIAL	0.000E+00		2.740E+08				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	2.734E+07	1.284E+11	2.466E+08	2.055E+05	5.877E+04	1.200E-05	3.561E+07
2	0.5000 h	0.000E+00	0.000E+00	8.636E+06	9.738E+09	1.870E+07	3.116E+04	6.160E+04	1.257E-05	3.612E+07
3	1.5000 h	0.000E+00	0.000E+00	8.578E+03	4.492E+09	8.624E+06	2.396E+03	5.361E+04	1.094E-05	2.069E+08
4	1.8333 h	0.000E+00	0.000E+00	8.559E+02	4.021E+06	7.720E+03	6.433E+00	5.056E+04	1.032E-05	6.248E+07
5	2.0000 h	0.000E+00	0.000E+00	2.704E+02	3.049E+05	5.854E+02	9.756E-01	4.910E+04	1.002E-05	2.989E+07
6	2.5000 h	0.000E+00	0.000E+00	2.700E+02	4.864E+05	0.000E+00	0.000E+00	4.496E+04	9.179E-06	8.460E+07
7	4.0000 h	0.000E+00	0.000E+00	2.691E+02	1.456E+06	0.000E+00	0.000E+00	3.454E+04	7.051E-06	2.134E+08
8	8.0000 h	0.000E+00	0.000E+00	2.665E+02	3.856E+06	0.000E+00	0.000E+00	1.709E+04	3.489E-06	3.572E+08
9	24.0000 h	0.000E+00	0.000E+00	2.563E+02	1.505E+07	0.000E+00	0.000E+00	1.026E+03	2.094E-07	3.290E+08
10	96.0000 h	0.000E+00	0.000E+00	2.152E+02	6.095E+07	0.000E+00	0.000E+00	3.256E-03	6.646E-13	2.100E+07
11	720.0000 h	0.000E+00	0.000E+00	4.717E+01	2.488E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	6.666E+01
Xe-131m	TOTALS		0.000E+00		1.430E+11	2.739E+08				1.376E+09

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

STEP	TIME	N/A CURRENT uCi	N/A INTEGRD uCi-sec	FHB CURRENT uCi	FHB INTEGRD uCi-sec	RELEASED uCi	AVERAGE RELEASE uCi/sec	CONTROL ROOM CURRENT uCi	CONTROL ROOM CURRENT uCi/cc	CONTROL ROOM INTEGRD uCi-sec
Xe-133m	INITIAL	0.000E+00		5.240E+08				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	5.210E+07	2.453E+11	4.710E+08	3.925E+05	1.121E+05	2.288E-05	6.794E+07
2	0.5000 h	0.000E+00	0.000E+00	1.643E+07	1.854E+10	3.560E+07	5.934E+04	1.172E+05	2.393E-05	6.880E+07
3	1.5000 h	0.000E+00	0.000E+00	1.614E+04	8.531E+09	1.638E+07	4.550E+03	1.010E+05	2.061E-05	3.918E+08
4	1.8333 h	0.000E+00	0.000E+00	1.605E+03	7.557E+06	1.451E+04	1.209E+01	9.487E+04	1.937E-05	1.175E+08
5	2.0000 h	0.000E+00	0.000E+00	5.061E+02	5.712E+05	1.097E+03	1.828E+00	9.196E+04	1.877E-05	5.604E+07
6	2.5000 h	0.000E+00	0.000E+00	5.027E+02	9.079E+05	0.000E+00	0.000E+00	8.377E+04	1.710E-05	1.580E+08
7	4.0000 h	0.000E+00	0.000E+00	4.929E+02	2.688E+06	0.000E+00	0.000E+00	6.332E+04	1.293E-05	3.946E+08
8	8.0000 h	0.000E+00	0.000E+00	4.676E+02	6.914E+06	0.000E+00	0.000E+00	3.002E+04	6.128E-06	6.425E+08
9	24.0000 h	0.000E+00	0.000E+00	3.786E+02	2.428E+07	0.000E+00	0.000E+00	1.516E+03	3.095E-07	5.499E+08
10	96.0000 h	0.000E+00	0.000E+00	1.465E+02	6.337E+07	0.000E+00	0.000E+00	2.219E-03	4.529E-13	2.926E+07
11	720.0000 h	0.000E+00	0.000E+00	3.909E-02	3.999E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.280E+01
Xe-133m	TOTALS		0.000E+00		2.725E+11	5.230E+08				2.476E+09
Xe-133	INITIAL	0.000E+00		2.810E+10				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	2.801E+09	1.317E+13	2.528E+10	2.107E+07	6.023E+06	1.229E-03	3.649E+09
2	0.5000 h	0.000E+00	0.000E+00	8.843E+08	9.975E+11	1.915E+09	3.192E+06	6.309E+06	1.288E-03	3.700E+09
3	1.5000 h	0.000E+00	0.000E+00	8.758E+05	4.598E+11	8.828E+08	2.452E+05	5.474E+06	1.117E-03	2.116E+10
4	1.8333 h	0.000E+00	0.000E+00	8.730E+04	4.104E+08	7.879E+05	6.566E+02	5.158E+06	1.053E-03	6.377E+09
5	2.0000 h	0.000E+00	0.000E+00	2.756E+04	3.109E+07	5.969E+04	9.949E+01	5.006E+06	1.022E-03	3.049E+09
6	2.5000 h	0.000E+00	0.000E+00	2.749E+04	4.955E+07	0.000E+00	0.000E+00	4.578E+06	9.345E-04	8.620E+09
7	4.0000 h	0.000E+00	0.000E+00	2.727E+04	1.478E+08	0.000E+00	0.000E+00	3.501E+06	7.147E-04	2.168E+10
8	8.0000 h	0.000E+00	0.000E+00	2.668E+04	3.884E+08	0.000E+00	0.000E+00	1.712E+06	3.495E-04	3.601E+10
9	24.0000 h	0.000E+00	0.000E+00	2.447E+04	1.472E+09	0.000E+00	0.000E+00	9.794E+04	1.999E-05	3.250E+10
10	96.0000 h	0.000E+00	0.000E+00	1.654E+04	5.249E+09	0.000E+00	0.000E+00	2.503E-01	5.109E-11	1.972E+09
11	720.0000 h	0.000E+00	0.000E+00	5.357E+02	1.050E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.037E+03
Xe-133	TOTALS		0.000E+00		1.464E+13	2.808E+10				1.387E+11
Xe-135m	INITIAL	0.000E+00		1.650E+05				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	6.731E+03	5.934E+07	1.139E+05	9.494E+01	1.803E+01	3.680E-09	1.251E+04
2	0.5000 h	0.000E+00	0.000E+00	1.364E+03	2.017E+06	3.872E+03	6.453E+00	1.197E+01	2.444E-09	8.856E+03
3	1.5000 h	0.000E+00	0.000E+00	1.122E-01	5.139E+05	9.866E+02	2.741E-01	8.850E-01	1.807E-10	1.470E+04
4	1.8333 h	0.000E+00	0.000E+00	5.846E-03	4.271E+01	8.200E-02	6.833E-05	4.163E-01	8.499E-11	7.376E+02
5	2.0000 h	0.000E+00	0.000E+00	1.422E-03	1.871E+00	3.592E-03	5.987E-06	3.025E-01	6.176E-11	2.131E+02
6	2.5000 h	0.000E+00	0.000E+00	8.496E-04	1.938E+00	0.000E+00	0.000E+00	1.525E-01	3.113E-11	3.813E+02
7	4.0000 h	0.000E+00	0.000E+00	5.694E-04	3.555E+00	0.000E+00	0.000E+00	7.375E-02	1.505E-11	5.416E+02
8	8.0000 h	0.000E+00	0.000E+00	3.722E-04	6.663E+00	0.000E+00	0.000E+00	2.405E-02	4.909E-12	6.366E+02
9	24.0000 h	0.000E+00	0.000E+00	6.952E-05	1.039E+01	0.000E+00	0.000E+00	2.802E-04	5.719E-14	3.075E+02
10	96.0000 h	0.000E+00	0.000E+00	3.657E-08	2.385E+00	0.000E+00	0.000E+00	5.572E-13	1.137E-22	3.625E+00
11	720.0000 h	0.000E+00	0.000E+00	1.396E-36	1.255E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.209E-09
Xe-135m	TOTALS		0.000E+00		6.187E+07	1.188E+05				3.898E+04

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

		N/A		FHB			AVERAGE	----------CONTROL ROOM----------		
		CURRENT	INTEGRD	CURRENT	INTEGRD	RELEASED	RELEASE	CURRENT	CURRENT	INTEGRD
STEP	TIME	uCi	uCi-sec	uCi	uCi-sec	uCi	uCi/sec	uCi	uCi/cc	uCi-sec
Xe-135	INITIAL	0.000E+00		5.430E+07				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	5.287E+06	2.525E+10	4.848E+07	4.040E+04	1.142E+04	2.331E-06	6.945E+06
2	0.5000 h	0.000E+00	0.000E+00	1.650E+06	1.874E+09	3.598E+06	5.996E+03	1.182E+04	2.412E-06	6.971E+06
3	1.5000 h	0.000E+00	0.000E+00	1.522E+03	8.490E+08	1.630E+06	4.528E+02	9.562E+03	1.952E-06	3.831E+07
4	1.8333 h	0.000E+00	0.000E+00	1.482E+02	7.079E+05	1.359E+03	1.133E+00	8.799E+03	1.796E-06	1.101E+07
5	2.0000 h	0.000E+00	0.000E+00	4.625E+01	5.253E+04	1.009E+02	1.681E-01	8.441E+03	1.723E-06	5.171E+06
6	2.5000 h	0.000E+00	0.000E+00	4.452E+01	8.168E+04	0.000E+00	0.000E+00	7.451E+03	1.521E-06	1.428E+07
7	4.0000 h	0.000E+00	0.000E+00	3.972E+01	2.272E+05	0.000E+00	0.000E+00	5.125E+03	1.046E-06	3.356E+07
8	8.0000 h	0.000E+00	0.000E+00	2.930E+01	4.931E+05	0.000E+00	0.000E+00	1.889E+03	3.856E-07	4.669E+07
9	24.0000 h	0.000E+00	0.000E+00	8.673E+00	9.759E+05	0.000E+00	0.000E+00	3.488E+01	7.121E-09	2.676E+07
10	96.0000 h	0.000E+00	0.000E+00	3.623E-02	4.086E+05	0.000E+00	0.000E+00	5.510E-07	1.125E-16	5.034E+05
11	720.0000 h	0.000E+00	0.000E+00	8.705E-23	1.714E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.950E-03
Xe-135	TOTALS		0.000E+00		2.798E+10	5.371E+07				1.902E+08
I-131	INITIAL	0.000E+00		1.160E+08				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	1.157E+07	5.436E+10	1.044E+08	8.698E+04	2.487E+04	5.078E-06	1.507E+07
2	0.5000 h	0.000E+00	0.000E+00	3.654E+06	4.121E+09	7.912E+06	1.319E+04	2.606E+04	5.321E-06	1.528E+07
3	1.5000 h	0.000E+00	0.000E+00	3.625E+03	1.900E+09	3.648E+06	1.013E+03	2.266E+04	4.625E-06	8.752E+07
4	1.8333 h	0.000E+00	0.000E+00	3.616E+02	1.699E+06	3.262E+03	2.718E+00	2.136E+04	4.360E-06	2.640E+07
5	2.0000 h	0.000E+00	0.000E+00	1.142E+02	1.288E+05	2.473E+02	4.121E-01	2.074E+04	4.233E-06	1.263E+07
6	2.5000 h	0.000E+00	0.000E+00	1.140E+02	2.054E+05	0.000E+00	0.000E+00	1.898E+04	3.875E-06	3.573E+07
7	4.0000 h	0.000E+00	0.000E+00	1.134E+02	6.139E+05	0.000E+00	0.000E+00	1.456E+04	2.971E-06	9.002E+07
8	8.0000 h	0.000E+00	0.000E+00	1.118E+02	1.621E+06	0.000E+00	0.000E+00	7.171E+03	1.464E-06	1.502E+08
9	24.0000 h	0.000E+00	0.000E+00	1.055E+02	6.256E+06	0.000E+00	0.000E+00	4.223E+02	8.620E-08	1.373E+08
10	96.0000 h	0.000E+00	0.000E+00	8.147E+01	2.410E+07	0.000E+00	0.000E+00	1.233E-03	2.516E-13	8.589E+06
11	720.0000 h	0.000E+00	0.000E+00	8.660E+00	7.297E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.507E+01
I-131	TOTALS		0.000E+00		6.049E+10	1.159E+08				5.787E+08
I-132	INITIAL	0.000E+00		6.020E+07				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	5.437E+06	2.733E+10	5.248E+07	4.373E+04	1.191E+04	2.431E-06	7.335E+06
2	0.5000 h	0.000E+00	0.000E+00	1.634E+06	1.898E+09	3.644E+06	6.074E+03	1.186E+04	2.422E-06	7.132E+06
3	1.5000 h	0.000E+00	0.000E+00	1.204E+03	8.149E+08	1.565E+06	4.346E+02	7.686E+03	1.569E-06	3.460E+07
4	1.8333 h	0.000E+00	0.000E+00	1.087E+02	5.465E+05	1.049E+03	8.743E-01	6.561E+03	1.339E-06	8.530E+06
5	2.0000 h	0.000E+00	0.000E+00	3.267E+01	3.795E+04	7.286E+01	1.214E-01	6.062E+03	1.237E-06	3.785E+06
6	2.5000 h	0.000E+00	0.000E+00	2.810E+01	5.459E+04	0.000E+00	0.000E+00	4.781E+03	9.759E-07	9.713E+06
7	4.0000 h	0.000E+00	0.000E+00	1.788E+01	1.221E+05	0.000E+00	0.000E+00	2.345E+03	4.788E-07	1.847E+07
8	8.0000 h	0.000E+00	0.000E+00	5.356E+00	1.496E+05	0.000E+00	0.000E+00	3.511E+02	7.167E-08	1.512E+07
9	24.0000 h	0.000E+00	0.000E+00	4.313E-02	6.347E+04	0.000E+00	0.000E+00	1.763E-01	3.600E-11	2.661E+06
10	96.0000 h	0.000E+00	0.000E+00	1.626E-11	5.152E+02	0.000E+00	0.000E+00	2.514E-16	5.133E-26	1.337E+03
11	720.0000 h	0.000E+00	0.000E+00	0.000E+00	1.943E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.907E-12
I-132	TOTALS		0.000E+00		3.004E+10	5.768E+07				1.073E+08

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

		N/A		FHB			AVERAGE	----------CONTROL ROOM----------		
		CURRENT	INTEGRD	CURRENT	INTEGRD	RELEASED	RELEASE	CURRENT	CURRENT	INTEGRD
STEP	TIME	uCi	uCi-sec	uCi	uCi-sec	uCi	uCi/sec	uCi	uCi/cc	uCi-sec
I-133	INITIAL	0.000E+00		7.430E+06				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	7.338E+05	3.471E+09	6.664E+06	5.553E+03	1.580E+03	3.226E-07	9.591E+05
2	0.5000 h	0.000E+00	0.000E+00	2.306E+05	2.608E+08	5.008E+05	8.346E+02	1.648E+03	3.363E-07	9.685E+05
3	1.5000 h	0.000E+00	0.000E+00	2.221E+02	1.194E+08	2.293E+05	6.368E+01	1.391E+03	2.839E-07	5.453E+06
4	1.8333 h	0.000E+00	0.000E+00	2.193E+01	1.037E+05	1.992E+02	1.660E-01	1.298E+03	2.650E-07	1.613E+06
5	2.0000 h	0.000E+00	0.000E+00	6.892E+00	7.796E+03	1.497E+01	2.495E-02	1.254E+03	2.560E-07	7.657E+05
6	2.5000 h	0.000E+00	0.000E+00	6.778E+00	1.230E+04	0.000E+00	0.000E+00	1.131E+03	2.309E-07	2.145E+06
7	4.0000 h	0.000E+00	0.000E+00	6.448E+00	3.570E+04	0.000E+00	0.000E+00	8.295E+02	1.693E-07	5.251E+06
8	8.0000 h	0.000E+00	0.000E+00	5.643E+00	8.693E+04	0.000E+00	0.000E+00	3.628E+02	7.406E-08	8.127E+06
9	24.0000 h	0.000E+00	0.000E+00	3.311E+00	2.519E+05	0.000E+00	0.000E+00	1.328E+01	2.710E-09	6.087E+06
10	96.0000 h	0.000E+00	0.000E+00	3.006E-01	3.252E+05	0.000E+00	0.000E+00	4.557E-06	9.303E-16	2.312E+05
11	720.0000 h	0.000E+00	0.000E+00	2.799E-10	3.247E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.936E-02
I-133	TOTALS		0.000E+00		3.852E+09	7.394E+06				3.160E+07
I-135	INITIAL	0.000E+00		5.060E+03				0.000E+00		
1	0.3333 h	0.000E+00	0.000E+00	4.879E+02	2.346E+06	4.504E+03	3.753E+00	1.056E+00	2.155E-10	6.431E+02
2	0.5000 h	0.000E+00	0.000E+00	1.515E+02	1.726E+05	3.314E+02	5.523E-01	1.087E+00	2.219E-10	6.429E+02
3	1.5000 h	0.000E+00	0.000E+00	1.359E-01	7.767E+04	1.491E+02	4.142E-02	8.551E-01	1.745E-10	3.477E+03
4	1.8333 h	0.000E+00	0.000E+00	1.310E-02	6.298E+01	1.209E-01	1.008E-04	7.793E-01	1.591E-10	9.799E+02
5	2.0000 h	0.000E+00	0.000E+00	4.068E-03	4.634E+00	8.897E-03	1.483E-05	7.440E-01	1.519E-10	4.569E+02
6	2.5000 h	0.000E+00	0.000E+00	3.860E-03	7.134E+00	0.000E+00	0.000E+00	6.474E-01	1.321E-10	1.250E+03
7	4.0000 h	0.000E+00	0.000E+00	3.298E-03	1.929E+01	0.000E+00	0.000E+00	4.265E-01	8.705E-11	2.858E+03
8	8.0000 h	0.000E+00	0.000E+00	2.168E-03	3.879E+01	0.000E+00	0.000E+00	1.401E-01	2.860E-11	3.704E+03
9	24.0000 h	0.000E+00	0.000E+00	4.050E-04	6.054E+01	0.000E+00	0.000E+00	1.632E-03	3.332E-13	1.791E+03
10	96.0000 h	0.000E+00	0.000E+00	2.130E-07	1.390E+01	0.000E+00	0.000E+00	3.246E-12	6.627E-22	2.112E+01
11	720.0000 h	0.000E+00	0.000E+00	8.132E-36	7.313E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.200E-08
I-135	TOTALS		0.000E+00		2.596E+06	4.984E+03				1.582E+04

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	EDE DOSE RATE mrem/h	CEDE DOSE mrem	TEDE DOSE mrem
Kr-85m										
0.3333 h	3.04E-05	0.00E+00	3.04E-05	1.47E-06	0.00E+00	1.47E-06	3.55E-08	2.09E-07	0.00E+00	3.55E-08
0.5000 h	2.21E-06	0.00E+00	2.21E-06	1.07E-07	0.00E+00	1.07E-07	3.52E-08	2.13E-07	0.00E+00	3.52E-08
1.5000 h	9.81E-07	0.00E+00	9.81E-07	4.74E-08	0.00E+00	4.74E-08	1.85E-07	1.60E-07	0.00E+00	1.85E-07
1.8333 h	7.58E-10	0.00E+00	7.58E-10	3.66E-11	0.00E+00	3.66E-11	5.05E-08	1.43E-07	0.00E+00	5.05E-08
2.0000 h	5.50E-11	0.00E+00	5.50E-11	2.66E-12	0.00E+00	2.66E-12	2.32E-08	1.36E-07	0.00E+00	2.32E-08
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.25E-08	1.15E-07	0.00E+00	6.25E-08
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.36E-07	7.04E-08	0.00E+00	1.36E-07
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.57E-07	1.89E-08	0.00E+00	1.57E-07
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.74E-08	9.94E-11	0.00E+00	5.74E-08
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.82E-10	5.46E-21	0.00E+00	1.82E-10
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.65E-21	0.00E+00	0.00E+00	6.65E-21
TOTALS	3.36E-05	0.00E+00	3.36E-05	1.62E-06	0.00E+00	1.62E-06	7.43E-07		0.00E+00	7.43E-07
Kr-85										
0.3333 h	2.81E-01	0.00E+00	2.81E-01	1.36E-02	0.00E+00	1.36E-02	3.33E-04	1.98E-03	0.00E+00	3.33E-04
0.5000 h	2.13E-02	0.00E+00	2.13E-02	1.03E-03	0.00E+00	1.03E-03	3.38E-04	2.07E-03	0.00E+00	3.38E-04
1.5000 h	9.83E-03	0.00E+00	9.83E-03	4.75E-04	0.00E+00	4.75E-04	1.94E-03	1.81E-03	0.00E+00	1.94E-03
1.8333 h	8.82E-06	0.00E+00	8.82E-06	4.26E-07	0.00E+00	4.26E-07	5.86E-04	1.71E-03	0.00E+00	5.86E-04
2.0000 h	6.70E-07	0.00E+00	6.70E-07	3.24E-08	0.00E+00	3.24E-08	2.81E-04	1.66E-03	0.00E+00	2.81E-04
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.95E-04	1.52E-03	0.00E+00	7.95E-04
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.01E-03	1.17E-03	0.00E+00	2.01E-03
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.38E-03	5.86E-04	0.00E+00	3.38E-03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.17E-03	3.66E-05	0.00E+00	3.17E-03
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.26E-04	1.38E-10	0.00E+00	1.26E-04
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.19E-10	0.00E+00	0.00E+00	3.19E-10
TOTALS	3.12E-01	0.00E+00	3.12E-01	1.51E-02	0.00E+00	1.51E-02	1.30E-02		0.00E+00	1.30E-02
Xe-131m										
0.3333 h	4.44E-01	0.00E+00	4.44E-01	2.15E-02	0.00E+00	2.15E-02	5.27E-04	3.13E-03	0.00E+00	5.27E-04
0.5000 h	3.37E-02	0.00E+00	3.37E-02	1.63E-03	0.00E+00	1.63E-03	5.34E-04	3.28E-03	0.00E+00	5.34E-04
1.5000 h	1.55E-02	0.00E+00	1.55E-02	7.51E-04	0.00E+00	7.51E-04	3.06E-03	2.85E-03	0.00E+00	3.06E-03
1.8333 h	1.39E-05	0.00E+00	1.39E-05	6.72E-07	0.00E+00	6.72E-07	9.24E-04	2.69E-03	0.00E+00	9.24E-04
2.0000 h	1.05E-06	0.00E+00	1.05E-06	5.10E-08	0.00E+00	5.10E-08	4.42E-04	2.61E-03	0.00E+00	4.42E-04
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.25E-03	2.39E-03	0.00E+00	1.25E-03
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.16E-03	1.84E-03	0.00E+00	3.16E-03
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.28E-03	9.10E-04	0.00E+00	5.28E-03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.87E-03	5.46E-05	0.00E+00	4.87E-03
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.86E-04	1.73E-10	0.00E+00	1.86E-04
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.94E-10	0.00E+00	0.00E+00	3.94E-10
TOTALS	4.94E-01	0.00E+00	4.94E-01	2.39E-02	0.00E+00	2.39E-02	2.02E-02		0.00E+00	2.02E-02

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	EDE DOSE RATE mrem/h	CEDE DOSE mrem	TEDE DOSE mrem
Kr-85m										
0.3333 h	3.04E-05	0.00E+00	3.04E-05	1.47E-06	0.00E+00	1.47E-06	3.55E-08	2.09E-07	0.00E+00	3.55E-08
0.5000 h	2.21E-06	0.00E+00	2.21E-06	1.07E-07	0.00E+00	1.07E-07	3.52E-08	2.13E-07	0.00E+00	3.52E-08
1.5000 h	9.81E-07	0.00E+00	9.81E-07	4.74E-08	0.00E+00	4.74E-08	1.85E-07	1.60E-07	0.00E+00	1.85E-07
1.8333 h	7.58E-10	0.00E+00	7.58E-10	3.66E-11	0.00E+00	3.66E-11	5.05E-08	1.43E-07	0.00E+00	5.05E-08
2.0000 h	5.50E-11	0.00E+00	5.50E-11	2.66E-12	0.00E+00	2.66E-12	2.32E-08	1.36E-07	0.00E+00	2.32E-08
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.25E-08	1.15E-07	0.00E+00	6.25E-08
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.36E-07	7.04E-08	0.00E+00	1.36E-07
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.57E-07	1.89E-08	0.00E+00	1.57E-07
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.74E-08	9.94E-11	0.00E+00	5.74E-08
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.82E-10	5.46E-21	0.00E+00	1.82E-10
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.65E-21	0.00E+00	0.00E+00	6.65E-21
TOTALS	3.36E-05	0.00E+00	3.36E-05	1.62E-06	0.00E+00	1.62E-06	7.43E-07		0.00E+00	7.43E-07
Kr-85										
0.3333 h	2.81E-01	0.00E+00	2.81E-01	1.36E-02	0.00E+00	1.36E-02	3.33E-04	1.98E-03	0.00E+00	3.33E-04
0.5000 h	2.13E-02	0.00E+00	2.13E-02	1.03E-03	0.00E+00	1.03E-03	3.38E-04	2.07E-03	0.00E+00	3.38E-04
1.5000 h	9.83E-03	0.00E+00	9.83E-03	4.75E-04	0.00E+00	4.75E-04	1.94E-03	1.81E-03	0.00E+00	1.94E-03
1.8333 h	8.82E-06	0.00E+00	8.82E-06	4.26E-07	0.00E+00	4.26E-07	5.86E-04	1.71E-03	0.00E+00	5.86E-04
2.0000 h	6.70E-07	0.00E+00	6.70E-07	3.24E-08	0.00E+00	3.24E-08	2.81E-04	1.66E-03	0.00E+00	2.81E-04
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.95E-04	1.52E-03	0.00E+00	7.95E-04
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.01E-03	1.17E-03	0.00E+00	2.01E-03
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.38E-03	5.86E-04	0.00E+00	3.38E-03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.17E-03	3.66E-05	0.00E+00	3.17E-03
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.26E-04	1.38E-10	0.00E+00	1.26E-04
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.19E-10	0.00E+00	0.00E+00	3.19E-10
TOTALS	3.12E-01	0.00E+00	3.12E-01	1.51E-02	0.00E+00	1.51E-02	1.30E-02		0.00E+00	1.30E-02
Xe-131m										
0.3333 h	4.44E-01	0.00E+00	4.44E-01	2.15E-02	0.00E+00	2.15E-02	5.27E-04	3.13E-03	0.00E+00	5.27E-04
0.5000 h	3.37E-02	0.00E+00	3.37E-02	1.63E-03	0.00E+00	1.63E-03	5.34E-04	3.28E-03	0.00E+00	5.34E-04
1.5000 h	1.55E-02	0.00E+00	1.55E-02	7.51E-04	0.00E+00	7.51E-04	3.06E-03	2.85E-03	0.00E+00	3.06E-03
1.8333 h	1.39E-05	0.00E+00	1.39E-05	6.72E-07	0.00E+00	6.72E-07	9.24E-04	2.69E-03	0.00E+00	9.24E-04
2.0000 h	1.05E-06	0.00E+00	1.05E-06	5.10E-08	0.00E+00	5.10E-08	4.42E-04	2.61E-03	0.00E+00	4.42E-04
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.25E-03	2.39E-03	0.00E+00	1.25E-03
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.16E-03	1.84E-03	0.00E+00	3.16E-03
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.28E-03	9.10E-04	0.00E+00	5.28E-03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.87E-03	5.46E-05	0.00E+00	4.87E-03
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.86E-04	1.73E-10	0.00E+00	1.86E-04
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.94E-10	0.00E+00	0.00E+00	3.94E-10
TOTALS	4.94E-01	0.00E+00	4.94E-01	2.39E-02	0.00E+00	2.39E-02	2.02E-02		0.00E+00	2.02E-02

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	EDE DOSE RATE mrem/h	CEDE DOSE mrem	TEDE DOSE mrem
Xe-133m										
0.3333 h	2.99E+00	0.00E+00	2.99E+00	1.45E-01	0.00E+00	1.45E-01	3.54E-03	2.10E-02	0.00E+00	3.54E-03
0.5000 h	2.26E-01	0.00E+00	2.26E-01	1.09E-02	0.00E+00	1.09E-02	3.59E-03	2.20E-02	0.00E+00	3.59E-03
1.5000 h	1.04E-01	0.00E+00	1.04E-01	5.03E-03	0.00E+00	5.03E-03	2.04E-02	1.90E-02	0.00E+00	2.04E-02
1.8333 h	9.22E-05	0.00E+00	9.22E-05	4.46E-06	0.00E+00	4.46E-06	6.12E-03	1.78E-02	0.00E+00	6.12E-03
2.0000 h	6.97E-06	0.00E+00	6.97E-06	3.37E-07	0.00E+00	3.37E-07	2.92E-03	1.73E-02	0.00E+00	2.92E-03
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.24E-03	1.57E-02	0.00E+00	8.24E-03
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.06E-02	1.19E-02	0.00E+00	2.06E-02
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.35E-02	5.64E-03	0.00E+00	3.35E-02
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.87E-02	2.85E-04	0.00E+00	2.87E-02
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.15E-04	4.16E-10	0.00E+00	9.15E-04
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.93E-10	0.00E+00	0.00E+00	8.93E-10
TOTALS	3.32E+00	0.00E+00	3.32E+00	1.61E-01	0.00E+00	1.61E-01	1.29E-01		0.00E+00	1.29E-01
Xe-133										
0.3333 h	1.83E+02	0.00E+00	1.83E+02	8.82E+00	0.00E+00	8.82E+00	2.16E-01	1.29E+00	0.00E+00	2.16E-01
0.5000 h	1.38E+01	0.00E+00	1.38E+01	6.68E-01	0.00E+00	6.68E-01	2.19E-01	1.35E+00	0.00E+00	2.19E-01
1.5000 h	6.38E+00	0.00E+00	6.38E+00	3.08E-01	0.00E+00	3.08E-01	1.25E+00	1.17E+00	0.00E+00	1.25E+00
1.8333 h	5.69E-03	0.00E+00	5.69E-03	2.75E-04	0.00E+00	2.75E-04	3.78E-01	1.10E+00	0.00E+00	3.78E-01
2.0000 h	4.31E-04	0.00E+00	4.31E-04	2.08E-05	0.00E+00	2.08E-05	1.81E-01	1.07E+00	0.00E+00	1.81E-01
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.11E-01	9.77E-01	0.00E+00	5.11E-01
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.29E+00	7.47E-01	0.00E+00	1.29E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.13E+00	3.65E-01	0.00E+00	2.13E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.93E+00	2.09E-02	0.00E+00	1.93E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.01E-02	5.34E-08	0.00E+00	7.01E-02
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.19E-07	0.00E+00	0.00E+00	1.19E-07
TOTALS	2.03E+02	0.00E+00	2.03E+02	9.80E+00	0.00E+00	9.80E+00	8.18E+00		0.00E+00	8.18E+00
Xe-135m										
0.3333 h	1.08E-02	0.00E+00	1.08E-02	5.20E-04	0.00E+00	5.20E-04	9.69E-06	5.03E-05	0.00E+00	9.69E-06
0.5000 h	3.66E-04	0.00E+00	3.66E-04	1.77E-05	0.00E+00	1.77E-05	6.86E-06	3.34E-05	0.00E+00	6.86E-06
1.5000 h	9.32E-05	0.00E+00	9.32E-05	4.50E-06	0.00E+00	4.50E-06	1.14E-05	2.47E-06	0.00E+00	1.14E-05
1.8333 h	7.75E-09	0.00E+00	7.75E-09	3.74E-10	0.00E+00	3.74E-10	5.72E-07	1.16E-06	0.00E+00	5.72E-07
2.0000 h	3.39E-10	0.00E+00	3.39E-10	1.64E-11	0.00E+00	1.64E-11	1.65E-07	8.44E-07	0.00E+00	1.65E-07
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.96E-07	4.26E-07	0.00E+00	2.96E-07
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.20E-07	2.06E-07	0.00E+00	4.20E-07
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.93E-07	6.71E-08	0.00E+00	4.93E-07
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.38E-07	7.82E-10	0.00E+00	2.38E-07
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.69E-09	1.55E-18	0.00E+00	1.69E-09
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.23E-18	0.00E+00	0.00E+00	2.23E-18
TOTALS	1.12E-02	0.00E+00	1.12E-02	5.42E-04	0.00E+00	5.42E-04	3.01E-05		0.00E+00	3.01E-05

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	EDE DOSE RATE mrem/h	CEDE DOSE mrem	TEDE DOSE mrem
Xe-135										
0.3333 h	2.68E+00	0.00E+00	2.68E+00	1.29E-01	0.00E+00	1.29E-01	3.15E-03	1.86E-02	0.00E+00	3.15E-03
0.5000 h	1.99E-01	0.00E+00	1.99E-01	9.60E-03	0.00E+00	9.60E-03	3.16E-03	1.93E-02	0.00E+00	3.16E-03
1.5000 h	9.00E-02	0.00E+00	9.00E-02	4.35E-03	0.00E+00	4.35E-03	1.74E-02	1.56E-02	0.00E+00	1.74E-02
1.8333 h	7.50E-05	0.00E+00	7.50E-05	3.63E-06	0.00E+00	3.63E-06	4.99E-03	1.44E-02	0.00E+00	4.99E-03
2.0000 h	5.57E-06	0.00E+00	5.57E-06	2.69E-07	0.00E+00	2.69E-07	2.34E-03	1.38E-02	0.00E+00	2.34E-03
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.47E-03	1.22E-02	0.00E+00	6.47E-03
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.52E-02	8.36E-03	0.00E+00	1.52E-02
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.12E-02	3.08E-03	0.00E+00	2.12E-02
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.21E-02	5.69E-05	0.00E+00	1.21E-02
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.37E-04	8.99E-13	0.00E+00	1.37E-04
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.44E-12	0.00E+00	0.00E+00	1.44E-12
TOTALS	2.97E+00	0.00E+00	2.97E+00	1.43E-01	0.00E+00	1.43E-01	8.61E-02		0.00E+00	8.61E-02
I-131										
0.3333 h	8.81E+00	1.50E+03	1.51E+03	4.26E-01	7.26E+01	7.30E+01	1.04E-02	6.20E-02	3.54E+01	3.54E+01
0.5000 h	6.68E-01	1.14E+02	1.15E+02	3.23E-02	5.50E+00	5.53E+00	1.06E-02	6.50E-02	3.59E+01	3.59E+01
1.5000 h	3.08E-01	5.25E+01	5.28E+01	1.49E-02	2.54E+00	2.55E+00	6.06E-02	5.65E-02	2.06E+02	2.06E+02
1.8333 h	2.75E-04	4.70E-02	4.72E-02	1.33E-05	2.27E-03	2.28E-03	1.83E-02	5.32E-02	6.21E+01	6.21E+01
2.0000 h	2.09E-05	3.56E-03	3.58E-03	1.01E-06	1.72E-04	1.73E-04	8.74E-03	5.17E-02	2.97E+01	2.97E+01
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.47E-02	4.73E-02	8.40E+01	8.40E+01
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.23E-02	3.63E-02	2.12E+02	2.12E+02
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.04E-01	1.79E-02	3.53E+02	3.53E+02
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.50E-02	1.05E-03	3.23E+02	3.23E+02
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.57E-03	3.07E-09	1.21E+01	1.21E+01
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.94E-09	0.00E+00	2.36E-05	2.36E-05
TOTALS	9.78E+00	1.67E+03	1.68E+03	4.73E-01	8.06E+01	8.11E+01	3.98E-01		1.35E+03	1.35E+03
I-132										
0.3333 h	2.72E+01	8.75E+00	3.59E+01	1.31E+00	4.23E-01	1.73E+00	3.11E-02	1.82E-01	2.00E-01	2.31E-01
0.5000 h	1.89E+00	6.07E-01	2.49E+00	9.11E-02	2.94E-02	1.20E-01	3.03E-02	1.81E-01	1.94E-01	2.24E-01
1.5000 h	8.09E-01	2.61E-01	1.07E+00	3.91E-02	1.26E-02	5.17E-02	1.47E-01	1.17E-01	9.42E-01	1.09E+00
1.8333 h	5.43E-04	1.75E-04	7.18E-04	2.62E-05	8.45E-06	3.47E-05	3.62E-02	1.00E-01	2.32E-01	2.68E-01
2.0000 h	3.77E-05	1.21E-05	4.98E-05	1.82E-06	5.87E-07	2.41E-06	1.61E-02	9.27E-02	1.03E-01	1.19E-01
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.12E-02	7.31E-02	2.64E-01	3.06E-01
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.84E-02	3.58E-02	5.03E-01	5.81E-01
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.42E-02	5.37E-03	4.12E-01	4.76E-01
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.13E-02	2.70E-06	7.24E-02	8.37E-02
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.41E-06	3.84E-21	2.18E-05	2.52E-05
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.24E-21	0.00E+00	2.08E-20	2.40E-20
TOTALS	2.98E+01	9.62E+00	3.95E+01	1.44E+00	4.65E-01	1.91E+00	4.56E-01		2.92E+00	3.38E+00

TRAILS12 -- Transport of Radioactive Material in Linear Systems, v1.0b
Unit 2 Fuel Handling Accident - CR, EAB & LPZ, 100 hours decay (2FHA_AST.TXT) ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	CEDE DOSE mrem	TEDE DOSE mrem	EDE DOSE mrem	EDE DOSE RATE mrem/h	CEDE DOSE mrem	TEDE DOSE mrem
I-133										
0.3333 h	9.07E-01	1.71E+01	1.80E+01	4.38E-02	8.24E-01	8.68E-01	1.07E-03	6.36E-03	4.01E-01	4.02E-01
0.5000 h	6.82E-02	1.28E+00	1.35E+00	3.29E-03	6.19E-02	6.52E-02	1.08E-03	6.63E-03	4.05E-01	4.06E-01
1.5000 h	3.12E-02	5.87E-01	6.18E-01	1.51E-03	2.84E-02	2.99E-02	6.09E-03	5.59E-03	2.28E+00	2.29E+00
1.8333 h	2.71E-05	5.10E-04	5.37E-04	1.31E-06	2.46E-05	2.59E-05	1.80E-03	5.22E-03	6.74E-01	6.76E-01
2.0000 h	2.04E-06	3.83E-05	4.03E-05	9.84E-08	1.85E-06	1.95E-06	8.55E-04	5.04E-03	3.20E-01	3.21E-01
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.40E-03	4.55E-03	8.96E-01	8.99E-01
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.87E-03	3.34E-03	2.19E+00	2.20E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.08E-03	1.46E-03	3.40E+00	3.41E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.80E-03	5.34E-05	2.54E+00	2.55E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.55E-04	1.83E-11	5.80E-02	5.81E-02
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.55E-11	0.00E+00	1.33E-08	1.33E-08
TOTALS	1.01E+00	1.89E+01	1.99E+01	4.86E-02	9.14E-01	9.63E-01	3.52E-02		1.32E+01	1.32E+01
I-135										
0.3333 h	1.66E-03	2.42E-03	4.08E-03	8.01E-05	1.17E-04	1.97E-04	1.94E-06	1.15E-05	5.65E-05	5.85E-05
0.5000 h	1.22E-04	1.78E-04	3.00E-04	5.89E-06	8.62E-06	1.45E-05	1.94E-06	1.18E-05	5.65E-05	5.84E-05
1.5000 h	5.49E-05	8.02E-05	1.35E-04	2.65E-06	3.88E-06	6.53E-06	1.05E-05	9.30E-06	3.06E-04	3.16E-04
1.8333 h	4.45E-08	6.51E-08	1.10E-07	2.15E-09	3.14E-09	5.29E-09	2.96E-06	8.47E-06	8.61E-05	8.91E-05
2.0000 h	3.27E-09	4.79E-09	8.06E-09	1.58E-10	2.31E-10	3.90E-10	1.38E-06	8.09E-06	4.02E-05	4.15E-05
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.78E-06	7.04E-06	1.10E-04	1.14E-04
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.63E-06	4.64E-06	2.51E-04	2.60E-04
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.12E-05	1.52E-06	3.26E-04	3.37E-04
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.41E-06	1.77E-08	1.57E-04	1.63E-04
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.83E-08	3.53E-17	1.11E-06	1.15E-06
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.07E-17	0.00E+00	1.48E-15	1.53E-15
TOTALS	1.83E-03	2.68E-03	4.52E-03	8.87E-05	1.30E-04	2.18E-04	4.78E-05		1.39E-03	1.44E-03
ALL NUCLIDES										
0.3333 h	2.26E+02	1.53E+03	1.75E+03	1.09E+01	7.38E+01	8.48E+01	2.67E-01	1.58E+00	3.60E+01	3.63E+01
0.5000 h	1.69E+01	1.16E+02	1.33E+02	8.18E-01	5.59E+00	6.41E+00	2.69E-01	1.65E+00	3.65E+01	3.68E+01
1.5000 h	7.74E+00	5.34E+01	6.11E+01	3.74E-01	2.58E+00	2.95E+00	1.51E+00	1.39E+00	2.09E+02	2.10E+02
1.8333 h	6.73E-03	4.76E-02	5.44E-02	3.25E-04	2.30E-03	2.63E-03	4.47E-01	1.30E+00	6.30E+01	6.34E+01
2.0000 h	5.06E-04	3.61E-03	4.12E-03	2.45E-05	1.74E-04	1.99E-04	2.12E-01	1.25E+00	3.01E+01	3.03E+01
2.5000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.96E-01	1.13E+00	8.51E+01	8.57E+01
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.47E+00	8.46E-01	2.14E+02	2.16E+02
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.38E+00	4.00E-01	3.57E+02	3.59E+02
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.09E+00	2.24E-02	3.25E+02	3.27E+02
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.52E-02	5.72E-08	1.22E+01	1.22E+01
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.28E-07	0.00E+00	2.36E-05	2.37E-05
TOTALS	2.51E+02	1.70E+03	1.95E+03	1.21E+01	8.20E+01	9.41E+01	9.31E+00		1.37E+03	1.38E+03

ATTACHMENT D-3

Beaver Valley Power Station, Unit Nos. 1 & 2
License Amendment Request Nos. 219 and 73

Attached is a Holtec International Report titled "Evaluation of Spent Fuel Assembly Drop Accidents in the Beaver Valley Power Station Reactor Core" (Non-Proprietary Version)

NON-PROPRIETARY VERSION

HOLTEC
INTERNATIONAL

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EVALUATION OF SPENT FUEL ASSEMBLY DROP ACCIDENTS IN THE BEAVER VALLEY POWER STATION REACTOR CORE

FOR

FIRST ENERGY NUCLEAR OPERATING COMPANY

Holtec Report No: HI-992343

Holtec Project No: 90844

Report Category: A

Report Class : SAFETY RELATED

HOLTEC INTERNATIONAL

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REVIEW AND CERTIFICATION LOG

DOCUMENT NAME :	EVALUATION OF SPENT FUEL ASSEMBLY DROP ACCIDENTS IN THE BEAVER VALLEY POWER STATION REACTOR CORE
HOLTEC DOCUMENT I.D. NUMBER :	992343
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REVISION BLOCK

REVISION NUMBER *	AUTHOR & DATE !!	REVIEWER & DATE !!	QA & DATE !!	APPROVED & DATE !	DIST. x
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QA AND ADMINISTRATIVE INFORMATION LOG

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Document No: **HI-992343**

CATEGORY: ☐ Generic ☒ **Project Specific**

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PREFACE

A fuel fragility analysis to determine the extent of rod breakage under various drop scenarios has been performed and documented in this report. The fuel assembly is assumed to be in an irradiated state, and for conservatism, modeled to simulate the actual geometry of Westinghouse 17×17 Optimized Fuel Assembly (OFA), which contains rods of the smallest diameter in the family of PWR fuel assemblies. The analysis was performed using the LS-DYNA code, which has been previously utilized by Holtec in numerous licensing basis analyses for impact and collision problems.

The analyzed drop scenarios are postulated to envelop all probabilistically significant drop events that could occur during fuel movement in the Beaver Valley Power Station reactor core. The results of the analysis, summarized in this report, show that the number of ruptured fuel rods could be up to 137. In summary, the fuel assembly drop analysis performed in this report has demonstrated that the postulated fuel drop accidents in the reactor core could lead to damage to fuel assemblies.

This is a nonproprietary report.

Revision Log

Revision 0 – Original Issue

Revision 1 – Significant changes have been made in Revision 1 of this report in accordance with Refs [9.6] and [9.9]. An additional scenario (Case 4) is analyzed in this revision. Some input data are also changed in this revision. The total weight of the dropped fuel assembly, which includes the attached grapples, inserted control rods, etc., is increased to 2500 lbs. In addition, the drop height is increased to 30 feet.

Revision 2 – Editorial errors have been corrected in Revision 2 per client's comments.

TABLE OF CONTENTS

1.0 PURPOSE AND SCOPE

Holtec International has been contracted by the FirstEnergy Nuclear Operating Company (FENOC) to perform spent fuel assembly damage evaluation under postulated drop events to support a revision to plant operating procedures and permit maintaining the containment in a non-isolated configuration. In accordance with the technical proposal [9.1], the fuel drop analysis is performed in two phases (i.e., Phase A and Phase B). The results of Phase A investigation were issued as an interim report [9.7]. This new report documents the results of both Phase A and Phase B analyses.

The objectives of this report include: (1) to define the bounding drop scenarios that warrant a fragility evaluation, (2) develop appropriate fuel drop analysis models, (3) perform a comprehensive analysis of the identified bounding drop scenarios with the developed models. By comparing the fuel rod sizes of potential dropped fuel assemblies [9.6], the Westinghouse 17×17 Optimized Fuel Assembly (OFA) has been identified to have the smallest rod size in the family of all PWR fuel assemblies [9.2], and its fuel rods are most vulnerable to drop accidents. Therefore, the modeling and analysis of fuel drop accidents were performed based on the geometry of the Westinghouse 17×17 OFA. In order to bound all drop events that may occur in the reactor core, four fuel assembly drop scenarios have been identified and analyzed in the report. Detailed description of the drop events is given in Section 6.1 of the report.

Fuel assembly drop events are transient, nonlinear problems involving a number of structural components in dynamic contacts. The finite-element method was used to conduct the numerical simulations for various drop events to obtain the conservative damage estimation. The impact damage was evaluated in terms of stress, strain and deformation in the structure members involved in the impact.

A commercial computer code developed by the Livermore Software Technology Corporation and QA validated by Holtec International, LS-DYNA [9.3], was used to numerically model the fuel handling accidents.

2.0 METHODOLOGY AND THEORETICAL BACKGROUND

The finite element method was used to carry out the impact analysis of postulated fuel drop accidents. Theoretically, the impact process of a drop event comprises the following three phases.

In the first phase, the impactor experiences a rigid body movement before it hits the target. During this process, the impactor accumulates kinetic energy while loosing its potential energy; other types of energy, if exist, remain constant. For the postulated drop events studied in this report, the impactor has a vertical trajectory traveling a specified distance before colliding with the target. The target is always stationary in this phase. The kinetic energy of the impactor can be described by two parameters: mass and velocity.

Phase two takes place from the contact initiation to the first full establishment of the contact. In this phase, energy redistributes between the impactor and the target. The kinetic energy of the impactor is absorbed by the target and the impactor in the form of internal energy due to the generated stress and deformation. To ensure the validity and accuracy of the analysis, all the possible contact interfaces and the corresponding contact conditions must be correctly defined. The contact interfaces may have a number of highly non-linear and time-dependent connectivity conditions, which are enforced and checked by the finite element model at every time step of the analysis.

The third phase begins after the full contact is developed. The shock wave generated by the impact propagates through the structural components of the impactor and the target. The type of the traveling wave depends on the material properties of the structural components, being of elastic or elasto-plastic nature. More importantly, permanent damage may be generated in the structural components of the target and impactor. During this phase the impactor may indent into or separate from the target, and additional but much weaker impacts may take place. However, the numerical analysis shall be considered complete when the velocity of the impactor is nullified or the failure criteria of the target material are exceeded.

A very important aspect of the impact analysis is the correctness of the material description. The elasto-plastic material behavior of the structural components involved in the impact can be defined as a bi-linear material through four constants: the elastic modulus, the yield stress, the rupture stress and the rupture strain. These constants, as they are obtained usually through laboratory tests, could deviate from the nominal values significantly when the deformation is applied in an impulsive manner or the material is irradiated for a long period of time. Both of these situations are present in the currently postulated drop events. It is known that a material subjected to an instantaneous deformation would increase its elastic stress limit but reduce the plastic strain before rupture. Moreover, the existence of radiation would make the behavior of an elasto-plastic material more brittle. If specific experimental data reflecting the variation of material properties in the drop event are not available, existing data in the technical literature pertinent to the static application of the load should be conservatively used in the numerical analysis to describe the elasto-plastic bi-linear material behavior.

The current numerical impact analysis takes a conservative approach to estimate the possible damage of the postulated drop accidents. In reality, the structural elements will dissipate a percentage of the initial impact kinetic energy throughout the deformation process. Due to the complexity of the analysis and the previously mentioned uncertainties related to the material description, the energy is intentionally channeled towards the structural elements to obtain an upper bound of the investigated damage.

In general, the impact analysis to assess the damage inflicted in the related structures is conducted in five sequential steps:

Step 1: description of the postulated drop event;

Step 2: identification and definition of the impactor and the target;

Step 3: impactor mass and impact velocity calculations;

Step 4: finite-element modeling of the impactor and the target;

Step 5: LS-DYNA numerical analysis and damage assessment.

3.0 ASSUMPTIONS

The following assumptions are used in the numerical analysis.

3.1 Assumptions related to the target and impactor:

3.1.1 Both the impactor and the target are entirely submerged during the collision, which is consistent with the postulated drop events.

3.1.2 The reactor core support is rigid and fixed. This assumption takes no credit of the energy absorption capacity of the reactor core support in a drop event, leading to a conservative damage estimate for the fuel assemblies involved in the drop accident.

3.1.3 The drag force on the leading face of the impactor that opposes the vertical motion of the impactor through the pool water is proportional to the square of the velocity.

3.1.4 The trajectory of the impactor is vertical, which is consistent with the postulated drop events.

3.1.5 Since a moving object in water always adjusts its position to reduce the drag force, it is assumed that all end-fitting pedestals of a vertically dropped fuel assembly simultaneously hit the target.

3.1.6 The frictional forces by which the fuel rods are attached to the guide tubes under normal conditions are conservatively neglected in vertical drop events to maximize the impact damage to the fuel rods.

3.2 Assumptions related to the material behavior:

3.2.1 The stainless steel (SA240-304) used to manufacture the fittings of the fuel assembly is considered as bi-linear elasto-plastic material as defined in Section 2.0. The material characteristics of the end fittings at a bounding temperature of 200 °F are obtained from Ref. [9.4].

3.2.2 The material of the guide tubes and fuel rods, Zircaloy-4, is also considered as bi-linear elasto-plastic. The material property is established based on the experimental data obtained under high strain rate and reasonable temperature conditions [9.5]. [redacted]

4.0 INPUT DATA

All fuel assembly geometrical dimensions, material specifications and estimated weights used in this analysis are collected from Ref. [9.2]. The material properties are taken from Refs. [9.4] and [9.5]. The total weight of the fuel assembly and the associated lifting tools is provided by Ref. [9.6]. Table 1 contains the geometric and material input data pertinent to the analysis.

Table 1. Geometric and Material Input Data for Fuel Drop Analysis

Fuel Assembly Data	Fuel Rod Outside Diameter (inch)	0.36
	Fuel Rod Inside Diameter (inch)	0.315
	Fuel Rod Length (inch)	151.56
	Number of Fuel Rods	264
	Fuel Rod Pitch (inch)	0.496
	Fuel Rod Array	17×17
	Fuel Pellet Weight Per Rod (lbs)	4.01
	Guide Tube Outside Diameter (inch)	0.474
	Guide Tube Inside Diameter (inch)	0.442
	Guide Tube Length (inch)	153.357
	Number of Guide Tubes	24
	Weight of Guide Tubes Per Assembly (lbs)	21.01
	Total Weight of Fuel Assembly	1373
	Total Weight of Fuel Assembly Plus Tools	2500
Material Properties	Yield Stress of SA240-304 (psi)	2.50×10^{4}
	Failure Stress of SA240-304 (psi)	7.10×10^{4}
	Young's Modulus of SA240-304 (psi)	2.76×10^{7}
	Failure Strain of SA240-304	0.38
	Yield Stress of Zircaloy-4 (psi)	
	Failure Stress of Zircaloy-4 (psi)	
	Young's Modulus of Zircaloy-4 (psi)	
	Failure Strain of Zircaloy-4	

Table 2 contains the input data pertinent to the analyzed drop scenarios. Note that the drop height for case 1 (i.e., fuel to reactor core support drop) is greater than that for cases 2 and 3 (i.e. fuel to fuel drops). The height difference is due to the height of the target fuel assembly, which is stored in reactor core and takes the hit of the dropped fuel assembly.

Table 2. Input Data for Analyzed Fuel Drop Scenarios

Drop Scenario	Drop Height (in)	Target	Impact Location on the Target	Impact Velocity (in/sec)
Case 1	360	Reactor core support (rigid)	----	
Case 2	200.235	Existing fuel in the core	Center of target fuel top fitting	
Case 3	200.235	Existing fuel in the core	Edge of target fuel top fitting	
Case 4	Rotational impact	Existing fuel in the core	Side of fuel rods	

5.0 COMPUTER CODES AND FILES

The Holtec QA validated LS-DYNA computer code [9.3], developed by Livermore Software Technology Corporation and QA validated by Holtec International, is used to model the drop events. This computer code has very sophisticated finite-element and material description libraries and can simulate various time-dependent contact conditions that normally arise in the structural components during the impact. In addition, MathCad 2000 is used to calculate the initial impact velocity of the fuel assembly.

The LS-DYNA input and output files for the analyzed accidents are stored in the Holtec network directory.

The numerical analysis is stored in the subdirectory F:\Projects\90884\JZ\Accident\FuelDrop. The results of analyzed drop scenarios are saved in four corresponding subdirectories (i.e., Case 1, Case 2, Case 3, and Case 4). Each subdirectory contains the LS-DYNA input file (*.DYN) corresponding to the analyzed drop event, and four time-history files (MATSUM- the impact velocity time-history, RCFORC- the impact force time-history, GLSTAT- the kinetic and internal energy time-histories and PLOT- the deformation time-history) generated in the numerical analysis.

The MathCad files for appendices A and B, as well as this report, are saved in the directory of F:\Projects\90884\JZ\Accident\Report.

The fuel drop accident analysis is performed under a PC Windows NT environment.

6.0 NUMERICAL ANALYSIS

6.1 Fuel Drop Event Description

Four impact scenarios specified in Refs. [9.6] and [9.9] are analyzed in this report. All drop events are postulated to occur during the handling of a Westinghouse 17×17 OFA fuel assembly. As pointed out in Section 1.0, the use of the above fuel assembly in the analysis bounds the drop events of other fuel assemblies in terms of damage to the fuel rods. Each of the drop scenarios is described below in detail.

6.1.1 Case 1

The fuel assembly is assumed to drop vertically with no initial velocity and travel through the stratum of water before impacting the rigid reactor core support. [redacted] The velocity of the dropped fuel assembly immediately before the impact is calculated to be [redacted] in Appendix A [redacted]. In addition, the rigid target (i.e., the reactor core support) assumption channels the entire impact energy into the fuel assembly, leading to a conservative damage evaluation for the dropped fuel assembly. In summary, evaluation of this drop scenario will envelop other drop scenarios in which the fuel assembly directly hits the reactor core support.

6.1.2 Case 2

A dropped fuel assembly may hit an existing fuel assembly stored in the reactor core. Note that the lower fitting of the dropped fuel assembly may impact any position of the upper fitting of the stationary fuel assembly stored in the reactor core. Case 2 simulates a scenario where the dropped fuel assembly exactly hits the center of the top fitting of the target fuel assembly. In this case, the top fitting of the target fuel assembly moves downward with little rotation as a result of the guide tubes deformation due to the "on-center" hit. The number of fuel rods that could be compressed by the upper and lower fittings is maximized in this drop scenario which therefore must be studied. Since the impact occurs at the elevation of the top fitting of the target fuel, the impact velocity is reduced to [redacted] due to the reduced drop height.

6.1.3 Case 3

The third drop case assumes that the dropped fuel assembly hits the edge of an existing fuel assembly stored in the reactor core. In this case, the fuel assembly fittings may rotate to some degree as a result of the "edge" impact. Compared with Case 2, some of the fuel rods could be compressed by the fittings more severely. To maximize the rotation of the end fittings, it is assumed that the only a half of width of the lower fitting pedestal of the dropped fuel assembly is in contact with the upper fitting of the target fuel assembly. The initial impact velocity of the dropped fuel assembly is same as that in Case 2.

6.1.3 Case 4

Case 4 simulates a scenario where a fuel assembly falls over and impacts the side of another standing fuel assembly with certain angular velocity. This scenario could occur after a vertically dropped fuel assembly hits the reactor core support or when a target fuel assembly is knocked and begins to "tipover" into other standing fuel assemblies in the reactor core. Therefore, Case 4 can be viewed as a potential event subsequent to any of the previous three scenarios. The simulation results can be used to estimate the total number of ruptured fuel rods after the "tipover" fuel assembly impacts one or multiple fuel assemblies. Note that the "tipover" fuel assembly does not impact the standing fuel assembly with a significant angular velocity after vertically hitting the reactor core support or a fuel assembly. The angular velocity immediately before the impact is conservatively calculated as [redacted] in Appendix B. In addition to the conservative angular velocity, the orientation of the impactor is chosen to be about 60 degrees relative to the reactor core support to maximize the impact when the target fuel assembly hits the reactor core support. For conservatism, the "tipover" fuel assembly is assumed to be rigid to maximize the damage to the target fuel assembly.

6.2 Finite-Element Model Description

[redacted]

[redacted]

7.0 SUMMARY OF RESULTS

The results of the LS-DYNA numerical analysis for the studied fuel handling accidents are summarized in the following subsections. LS-DYNA determines the failure of any structure element based on the specified failure stress and failure strain of the material. [redacted].

7.1 Case 1

For case 1 (i.e., the dropped fuel hits the reactor core support), Fig. 2 shows the velocity time history of the lower end fitting of the dropped fuel assembly. As shown in the figure, the first impact takes [redacted] after which the end fitting starts to bounce. [redacted].

7.2 Case 2

For case 2 (i.e., the center of the top fitting of a target fuel is hit), Fig. 7 shows the velocity time histories of the lower end fittings of the dropped fuel assembly and the target fuel assembly, represented by the solid line and the dashed line, respectively. [redacted]

[redacted].

7.3 Case 3

Case 3 is similar to case 2, but the first impact occurs at the edge of the top fitting of the target fuel assembly. As discussed in Section 6.1.3, the fuel assembly fittings may rotate to some degree as a result of the "edge" impact leading to more severe damage to the fuel rods close to the impact edge. [redacted].

7.4 Case 4

As discussed in Section 6.1.3, Case 4 can be viewed as an extended event of any of the previous three cases. Without repeating the vertical drop portion, simulation of Case 4 starts from the side impact between the "tipover" fuel assembly and a standing fuel assembly and ends after the impacted fuel assembly hits the reactor core support. [redacted].

8.0 CONCLUSIONS

A comprehensive finite element analysis has been performed to assess the damage to the spent fuel assembly resulting from various fuel assembly drop accidents postulated to occur in the Beaver Valley Power Station reactor core. The analysis was carried out with the commercial code LS-DYNA. It is found that the postulated drop events will result in up to 137 ruptured fuel rods.

9.0 REFERENCES

9.1 Technical Proposal, Holtec File No. H-3968, August 3, 1999.

9.2 Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long Term Isolation, Appendix 2A, Physical Descriptions of LWR Fuel Assemblies, US Department of Energy, December 1987.

9.3 LS-DYNA, Version 950, Livermore Software Technology Corporation, 1999.

9.4 ASME, "Boiler & Pressure Vessel Code," Section II, Part D – Material Properties, American Society of Mechanical Engineers, July 1, 1995.

9.5 Ramsey Chun et al., "Dynamic Impact Effects on Spent Fuel Assemblies," Lawrence Livermore National Laboratory, October 20, 1987.

9.6 Client's Comments on Rev. 0 of Holtec Report HI-992343, Feb. 4, 2000.

9.7 "Interim Report on the Evaluation of Spent Fuel Assembly Drop Accident in the Beaver Valley Power Station Reactor Core," Holtec Report HI-992324, Nov. 19, 1999.

9.8 "Beaver Valley Power Station Units 1 and 2 Data Supporting the Holtec Fuel Handling Analysis," Westinghouse, Jan. 31, 2000.

9.9 Comments on Fuel Handling Accident Analysis Document HI-992343 (Draft of Rev. 1), Beaver Valley Power Station, ND1MCM: 0072, May 30, 2000.

FIGURES

Fig. 1 Case 1: Finite Element Model

Fig. 2 Case 1: Velocity Time History – Lower End Fitting

Fig. 3 Case 1: Impact Force Time History

Fig. 4 Case I: Deformed Shape of the Dropped Fuel Assembly

Fig. 5 Case 1: Deformed Shape – Lower End Fitting

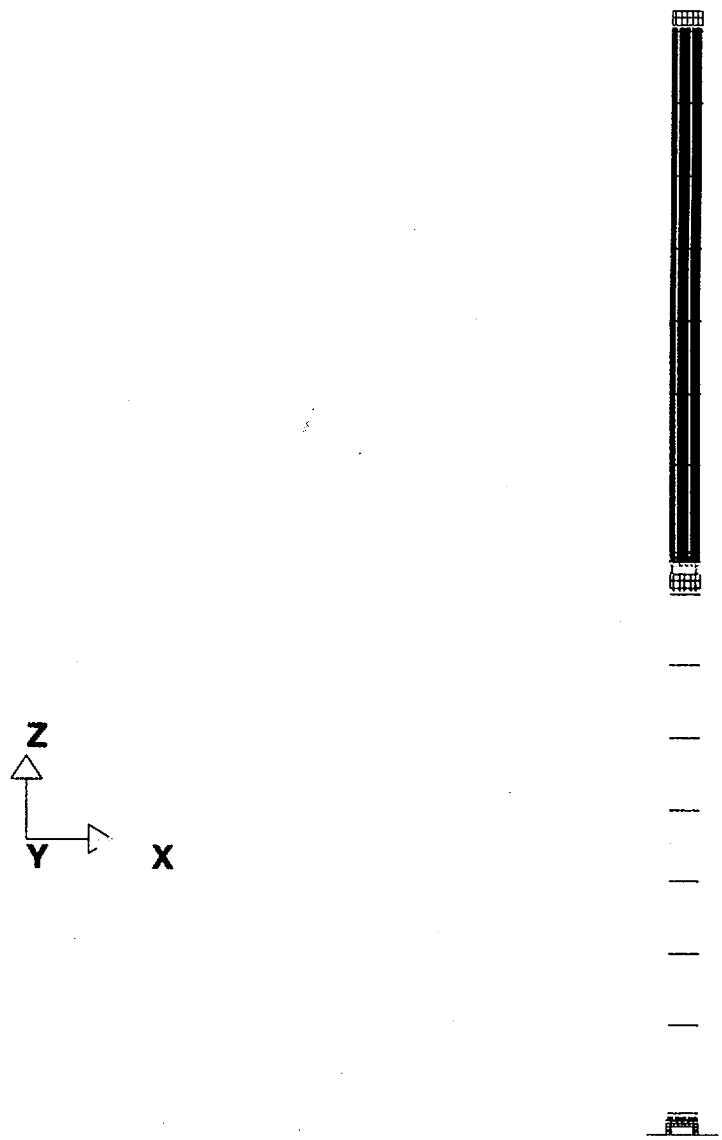

Fig. 6 Case 2: Finite Element Model

Fig. 7 Case 2: Impact Velocity Time History – Lower End Fittings

Fig. 8 Case 2: Impact Force Time History – Fuel to Fuel and Fuel to Floor

Fig. 9 Case 2: Deformed Shape of Fuel Assemlies

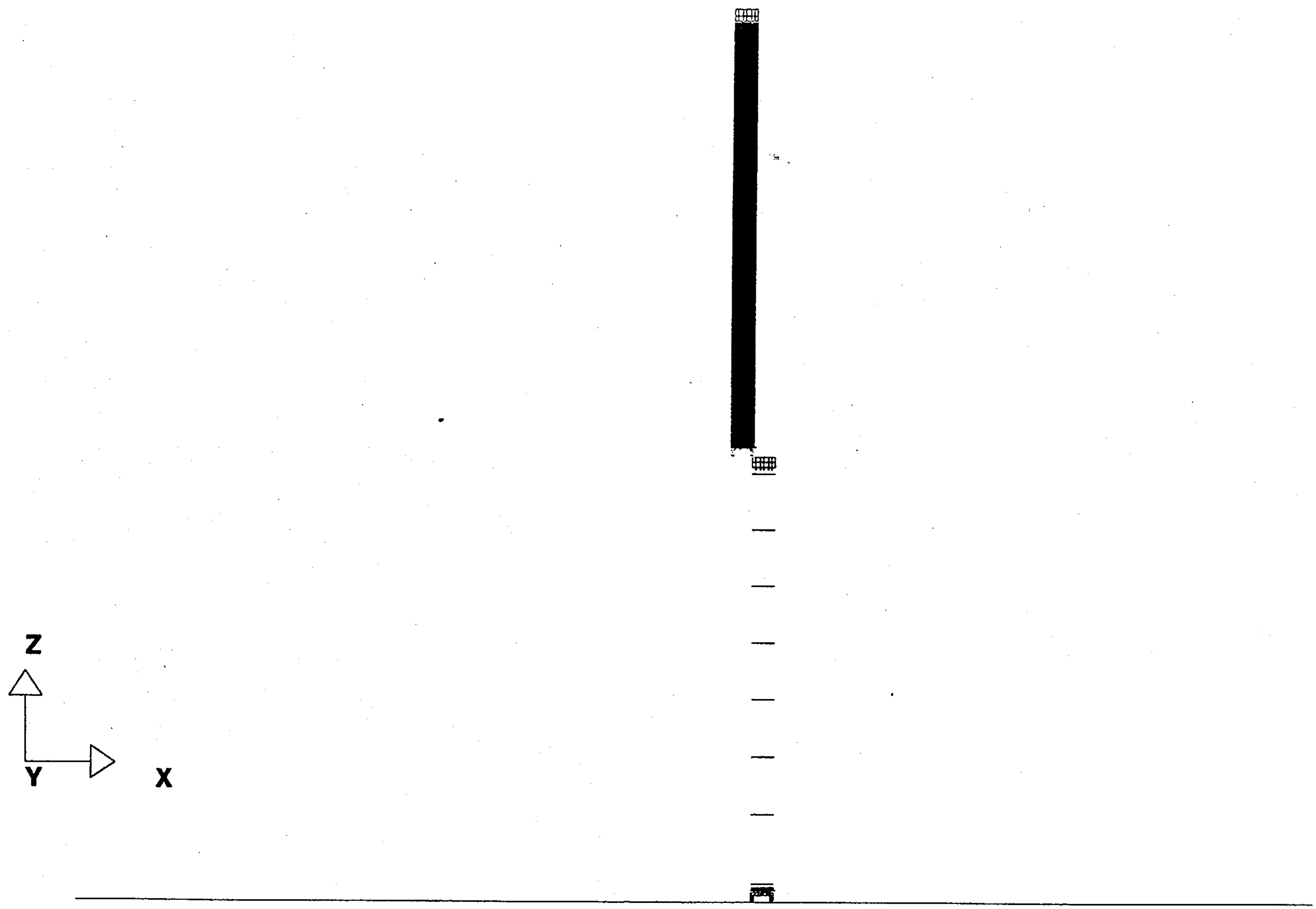

Fig. 10 Case 3: Finite Element Model

Fig. 11 Case 3: Snapshot of Deformed Fuel Assemblies (1)

Fig. 12 Case 3: Snapshot of Deformed Fuel Assemblies (2)

Fig. 13 Case 3: Snapshot of Deformed Fuel Assemblies (3)

Fig. 14 Case 3: Snapshot of Deformed Fuel Assemblies (4)

Fig. 15 Case 3: Snapshot of Deformed Fuel Assemblies (5)

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Fig. 16 Case 3: Deformed Fuel Assemblies – Lower End Fittings

Fig. 17 Case 3: Impact Force Time Histories of the Dropped Fuel Assembly

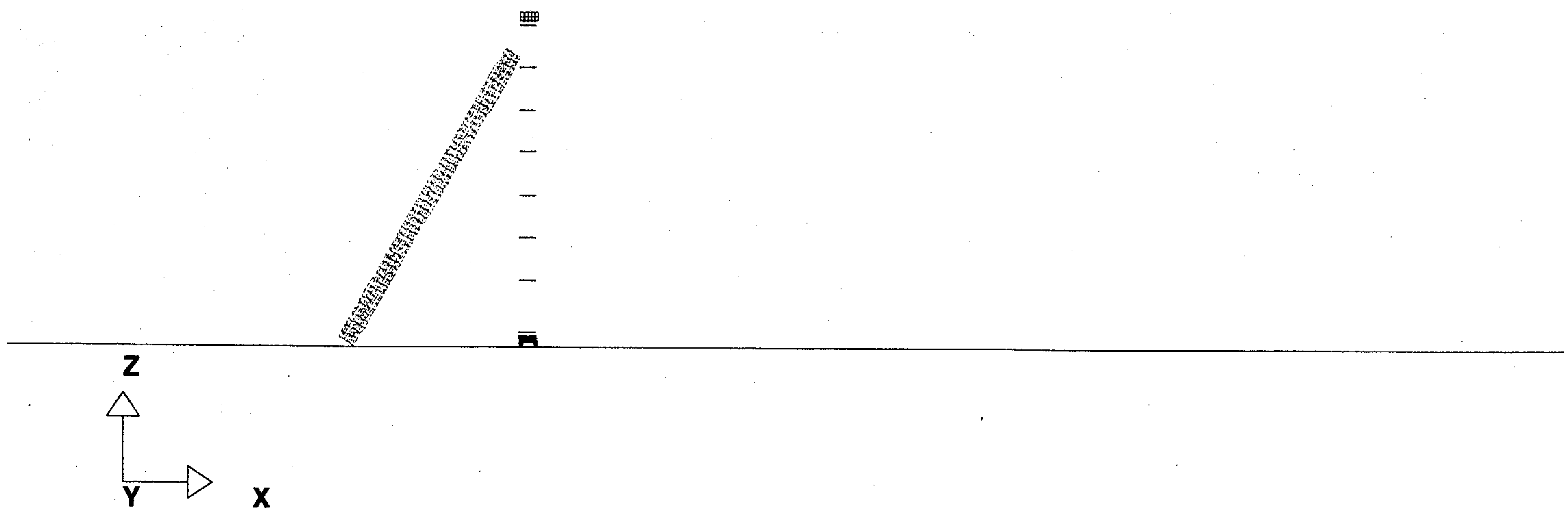

Fig. 18 Case 4: Finite Element Model

Fig. 19 Case 4: Snapshot of Fuel Assemblies During the Impact Event (1)

Fig. 20 Case 4: Snapshot of Fuel Assemblies During the Impact Event (2)

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Fig. 21 Case 4: Snapshot of Fuel Assemblies During the Impact Event (3)

Fig. 22 Case 4: Snapshot of Fuel Assemblies During the Impact Event (4)

Fig. 23 Case 4: Snapshot of Fuel Assemblies During the Impact Event (5)

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Appendix A: Impact Velocity Calculation for Fuel Assembly Drop Accidents

Part A: Theory

A.1 Drop of an Object in Water

A dropped object, such as a spent fuel assembly, is modeled as a single lumped mass under the influence of gravity in a drag inducing medium. The objective of the analysis is to calculate the impact velocity of the fuel assembly dropping through the water resulting from a fuel handling drop accident. The effects of virtual mass, gravity, and fluid drag are accounted for in the model. The virtual mass is assumed equal to the buoyant mass of the dropped object. The drag force is based on the exposed frontal area of the object. The governing equation for a body of mass subject to gravity and drag effects is

$$\left(M + M_v\right)\cdot\frac{dv}{dt} + \frac{C_D}{2}\cdot d_w\cdot A_D\cdot v^2 := \left(M - M_v\right)\cdot g \qquad \text{(A-1)}$$

where

M equals the dry mass of the object;
M_v equals the virtual mass of the object (buoyant mass);
C_D equals the effective drag coefficient due to all contributing effects;
d_w equals the mass density of the object;
A_D equals the area subject to drag forces;
v equals the velocity of the object;
g equals the gravitational acceleration.

If $\varepsilon = M_v/M$, then equation A-1 can be written as follows

$$\frac{dv}{dt} := \frac{1-\varepsilon}{1+\varepsilon}\cdot g - \frac{C_D\cdot d_w\cdot A_D\cdot v^2}{2\cdot M\cdot\left(1+\varepsilon\right)} \qquad \text{(A-2)}$$

The time derivative of velocity, dv/dt, is expressed as

$$\frac{dv}{dt} := v \cdot \frac{dv}{ds}$$

where

$$v := \frac{ds}{dt}$$

If the variable x is defined equal to v^2, then equation A-2 becomes

$$\frac{dx}{ds} + \lambda \cdot x := \beta \qquad \text{(A-3)}$$

where

$$\lambda := \frac{C_D \cdot d_w \cdot A_D}{M \cdot (1 + \varepsilon)} \quad ; \quad \beta := 2 \cdot \frac{1 - \varepsilon}{1 + \varepsilon} \cdot g$$

The constant λ is a measure of the drag forces relative to the inertia forces.

The solution to equation A-3, with initial condition $v = v_o$ at $s = 0$, is

$$x := \left(v_o^{\,2} - \frac{\beta}{\lambda} \right) \cdot e^{-\lambda \cdot s} + \frac{\beta}{\lambda} \qquad \text{(A-4)}$$

Part B: Calculation

B.0 INPUT DATA

The following parameters are general for all analyses:

Weight of fuel assembly and tools	$W := 2500 \cdot lbf$	[Ref.2]
Fuel assembly envelope	$D_{fuel} := 8.424in$	[Ref.1]
Mass density of water (200° F)	$d_w := 60.1 \cdot \frac{lb}{ft^3}$	
Mass density of fuel assembly (100% UO_2)	$d_f := 0.39 \cdot \frac{lb}{in^3}$	
Mass density of stainless steel	$d_{stl} := 490 \cdot \frac{lb}{ft^3}$	
Initial velocity of fuel assembly	$v_o := 0 \cdot \frac{in}{sec}$	

B.1 CALCULATION

The following calculation determines the velocity of the dropped fuel assembly immediately before the impact. Part A provides the theoretical background for this calculation. The nomenclature used herein is consistent with Part A.

The following parameters are defined in Part A.

$$\varepsilon := \frac{M_v}{M} \qquad \varepsilon := \frac{d_w \cdot V}{d_f \cdot V} \qquad \varepsilon := \frac{d_w}{d_f} \qquad \varepsilon = 0.089$$

Frontal area subject to drag force $A_D := D_{fuel} \cdot D_{fuel}$ $A_D = 71\,in^2$

For the drop of a fuel assembly to the floor or another fuel, there is no confinement of fluid, so the drag coefficient, C_D, equals 1.0.

Drag coefficient $C_D := 1.0$

B.1.1 Fuel Drop onto Reactor Core Floor

In this case, the drop height is: [Ref.2]

$$\lambda := \frac{C_D \cdot d_w \cdot A_D \cdot g}{W \cdot (1 + \varepsilon)} \qquad \text{(Eq. A-3)}$$

$$\beta := 2 \cdot \frac{(1 - \varepsilon)}{(1 + \varepsilon)} \cdot g \qquad \text{(Eq. A-3)}$$

$$x := \left(v_o^{\,2} - \frac{\beta}{\lambda}\right) \cdot \exp(-\lambda \cdot h) + \frac{\beta}{\lambda} \qquad \text{(Eq. A-4)}$$

$$v_{fl} := \sqrt{x}$$

The velocity of the dropped fuel assembly as it hits the floor is

B.1.2 Fuel Drop onto a Stored Fuel Assembly

In this case, the drop height (after subtracting the height of the fuel assembly) is: [Ref.2]

$$\lambda := \frac{C_D \cdot d_w \cdot A_D \cdot g}{W \cdot (1 + \varepsilon)} \qquad \text{(Eq. A-3)}$$

$$\beta := 2 \cdot \frac{(1 - \varepsilon)}{(1 + \varepsilon)} \cdot g \qquad \text{(Eq. A-3)}$$

$$x := \left(v_o^{\,2} - \frac{\beta}{\lambda}\right) \cdot \exp(-\lambda \cdot h) + \frac{\beta}{\lambda} \qquad \text{(Eq. A-4)}$$

$v_{f2} := \sqrt{x}$

The velocity of the dropped fuel assembly as it hits another fuel is

B2 REFERENCES

1. Characteristics of Spent Fuel, High_Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation, Appendix 2A, Physical Descriptions of LWR Fuel Assemblies, US Department of Energy, December 1987.

2. Project Input Data Provided by K.E. Halliday of Beaver Valley Nuclear Station Through Fax, Dated on January 4, 2000.

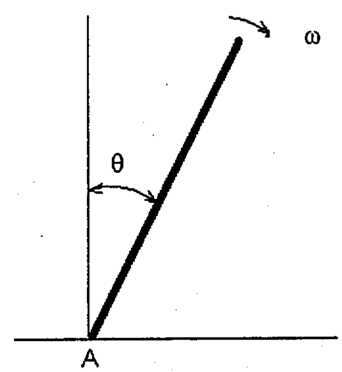
Appendix B

Calculation of Angular Impact Velocity of a Dropped Fuel Assembly

This appendix calculates the maximum angular impact velocity with which a fuel assembly hits another standing fuel assembly after a vertical drop to the reactor core at the Beaver Valley Power Station. The dropped fuel assembly is assumed to tip over with zero initial velocity trigged by some perturbation after the vertical drop. The angular velocity of the fuel assembly during the process of tipover reaches the maximum value when the fuel assembly rotates to the horizontal orientation. However, at any rotation angle, the fuel assembly may hit another standing fuel assembly stored in the reactor core and causes the latter to tipover. For conservatism, the maximum angular velocity is assumed to be the impact angular velocity.

The motion of a tipover fuel assembly with a weight of W is governed by the following differential equation

$$I_A \cdot \frac{d^2\theta}{dt^2} := W \cdot r \cdot \sin\theta$$



where, as shown in the following figure, I_A is the mass moment of inertia of the fuel assembly about the pivot point A, and θ is the rotation angle between a vertical line and the longitudinal axis of the fuel assembly.

Using the initial conditions, i.e., θ=0 and ω=dθ/dt=0 at t=0, the equation of motion can be integrated to give the angular velocity at any rotation angle.

$$\omega := \sqrt{\frac{2W \cdot r}{I_A} \cdot (1 - \cos\theta)}$$

When the fuel assembly rotates to the horizontal orientation (i.e., θ=90), the angular velocity reaches the maximum value.

$$\omega_{max} := \sqrt{\frac{2W \cdot r}{I_A}}$$

By taking the fuel assembly as a uniform bar, we have

$$I_A := \frac{ml^2}{3} \qquad r := \frac{l}{2}$$

Where, m and l are the mass and length of the fuel assembly, respectively. Substituting I_A and r into the angular velocity formula to yield

$$\omega_{max} := \sqrt{\frac{3g}{l}}$$

Per Ref. [9.2], $l :=$

$$\omega_{max} := \sqrt{\frac{3g}{l}} \qquad \omega_{max} =$$

Therefore, the maximum angular velocity at which the dropped fuel assembly could hit another standing fuel assembly in the reactor core is

ATTACHMENT D-1

Beaver Valley Power Station, Unit Nos. 1 & 2
License Amendment Request Nos. 219 and 73

Attached is an Affidavit Pursuant to 10 CFR 2.790

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797-0900
Fax (856) 797-0909

AFFIDAVIT PURSUANT TO 10CFR2.790

I, Scott H. Pellet, being duly sworn, depose and state as follows:

(1) I am an adjunct Project Manager for Holtec International and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the document entitled "Evaluation of Spent Fuel Assembly Drop Accidents in the Beaver Valley Power Station Reactor Core," Holtec Report HI-992343, revision 2. The proprietary material in this document is delineated by shaded text.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.790(a)(4), and 2.790(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

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b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b, 4.d, and 4.e, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

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(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed historical data and analytical results not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed using codes developed by Holtec International. Release of this information would improve a competitor's position without the competitor having to expend similar resources for the development of the database. A substantial effort has been expended by Holtec International to develop this information.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

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The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. Holtec International' s competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Scott H. Pellet, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 7th day of December, 2000.

Scott H. Pellet

Mr. Scott H. Pellet
Holtec International

Subscribed and sworn before me this 7th day of December 2000.

Maria C. Pepe