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May 1, 2000
L-00-048

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

Subject: Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
License Amendment Request No. 155

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company requests an amendment to the above license in the form of changes to the technical specifications.

The proposed amendment will revise the containment closure requirements specified in Technical Specification 3/4.9.4 titled "Containment Building Penetrations." The proposed amendment will also revise the requirements for the containment purge and exhaust radiation monitoring instrumentation contained in Technical Specification 3/4.3.3 titled "Monitoring Instrumentation - Radiation Monitoring." The requirements for the containment purge and exhaust isolation system contained in Technical Specification 3/4.9.9 titled "Containment Purge and Exhaust Isolation System" will also be revised by this proposed amendment. The proposed amendment also includes changes to the applicable Technical Specification Bases Sections, administrative, editorial and format changes.

The proposed technical specification changes for Unit No. 2 are presented in Attachment A. 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 is presented in Attachment C. Draft Updated Final Safety Analysis Report changes are presented in Attachment D.

This change is requested to be approved prior to the start of refueling outage 2R08 in September 2000.

These changes have been reviewed by the Beaver Valley review committees. The changes were determined to be safe and do not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis. An implementation period of up to 60 days is requested following the effective date of this amendment.

A001

If there are any questions concerning this matter, please contact Mr. B. F. Sepelak, Supervisor, Licensing at 412-393-5282.

Sincerely,

Lew W. Myers

c: Mr. D. S. Collins, 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)
Ms. Mary E. O'Reilly (FirstEnergy Legal Department)

I, Marc P. Pearson, being duly sworn, state that I am Director, Plant Services 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

Marc P. Pearson

Marc P. Pearson
Director, Plant Services - FENOC

STATE OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 15th day of May, 2000.

Tracey A Baczek

My Commission Expires:

Notarial Seal
Tracey A. Baczek, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Aug. 16, 2001
Member, Pennsylvania Association of Notaries

ATTACHMENT A

Beaver Valley Power Station, Unit No. 2
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(Proposed Wording)

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(Proposed wording)

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
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)	~~1~~ 2	~~6~~ (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

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

(2) With irradiated fuel in the storage pool.

(3) Above background.

(4) During movement of irradiated fuel.

ADD → (5) During movement of fuel assemblies within 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.

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 (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	~~6~~ ###
iii. Main Steam Discharge (2MSS-RQ101A, B & C)	S	R	M	1, 2, 3 & 4

ADD

During movement of fuel assemblies within containment.

(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, ; REPLACE WITH INSERT "A"

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

c. Each penetration (excluding the PAL) 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~~ REPLACE WITH INSERT "B"

2. REPLACE WITH INSERT "C" ~~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 ~~CORE ALTERATIONS or~~ movement of ~~irradiated~~ fuel assemblies within the containment. DELETE

ACTION:

DELETE DELETE

With the requirements of the above specification not satisfied, immediately suspend all operations involving ~~CORE ALTERATIONS or~~ movement of ~~irradiated~~ fuel assemblies in 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 ~~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.~~ : a. and REPLACE WITH INSERT "D"

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

a. Verifying the flow rate ~~through~~ to the 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~~, and~~ .

REPLACE WITH INSERT "E" ~~c. Testing the SLCRS per Specification 4.7.8.1 with the exception of item 4.7.8.1.c.2.~~

(Proposed Wording)

INSERT A

or both doors of the containment personnel air lock (PAL) may be open if:

1. At least one of the PAL doors is capable of being closed,

2. A designated individual is available to close at least one PAL door,

3. The PAL area is being exhausted to at least one OPERABLE filtered Supplemental Leak Collection and Release System (SLCRS) train with all doors (except for the air lock doors) to the PAL area closed[1], and

4. SR 4.9.4.4 has been satisfied with both PAL doors open; and

---

(1) Except for entry and exit.

INSERT B

or the penetration may be open if:

a. The penetration is capable of being closed by an isolation valve, blind flange, manual valve, or approved functional equivalent,

b. The maximum equivalent containment penetration opening size for the associated plant area is not exceeded,

c. A designated individual is available to close the penetration, and

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

---

(1) Except for entry and exit.

Attachment A
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INSERT C

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 SLCRS train.

INSERT D

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]

---

(1) Except for entry and exit.

INSERT E

4.9.4.3 The required portions of filtered SLCRS shall be demonstrated OPERABLE per Specification 4.7.8.1 with exception to item 4.7.8.1.c.2.

4.9.4.4 For areas required to be exhausted to filtered SLCRS (except for the containment), verify at least once per 7 days that filtered SLCRS can maintain the area at a negative pressure of ≤ -0.125 inches of water gauge with respect to atmospheric pressure. The verification shall establish the maximum equivalent containment penetration opening size for each applicable plant area.

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 ~~CORE ALTERATIONS or~~ movement of ~~irradiated~~ fuel assemblies within the containment.

DELETE

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 radiation signal from each of the containment radiation monitoring instrumentation channels.

DELETE

(Proposed wording)

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

DELETE

← 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 ~~trace amounts of~~ radioactive material within containment, will be filtered through HEPA filters and charcoal absorbers prior to discharge to the atmosphere. These

ADD

released through open containment penetrations, as the result of a fuel handling accident (FHA)

(Proposed Wording)

REPLACE WITH INSERT "F"

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

requirements 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.~~

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.~~

and open penetrations that meet the requirements of this specification,

FHA occurring inside containment.

ADD INSERT "G"

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 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.

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

(Proposed Wording)

INSERT F

2.34 fuel assemblies (617 fuel rods) being ruptured as a result of a FHA based upon the lack of containment pressurization potential while moving fuel assemblies within containment.

INSERT G

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.

A PAL door is considered capable of being closed when the following criteria are satisfied:

1. Administrative procedures have been established to:

    a. ensure that appropriate personnel are aware of the Open status of the containment during movement of fuel within the containment;

    b. ensure that an open air lock is capable of rapid closure (i.e., ≤ 30 minutes), with quick disconnect and removal capability for hoses, cables, ramps, and door seal protective covers; and

    c. ensure that an individual is designated and available to close the air lock following the evacuation that would occur in the event of an accident.

A containment penetration is considered capable of being closed when the following criteria are satisfied:

INSERT G (Continued)

1. Administrative procedures have been established to:

   a. ensure that appropriate personnel are aware of the Open status of the containment during movement of fuel within the containment;

   b. ensure that the containment penetration is capable of rapid closure (i.e., $\leq$ 30 minutes) by closing an isolation valve, manual valve, blind flange, or approved functional equivalent; and

   c. ensure that an individual is designated and available to close the containment penetration.

LCO 3.9.4.b.4 requires that SR 4.9.4.4 has been satisfied with both PAL doors open. This requirement is necessary to ensure that the opening of PAL will not adversely affect the ability of filtered SLCRS to maintain the PAL area at a negative pressure. LCO 3.9.4.c.1.b permits a containment penetration (excluding the PAL) to be open if the maximum equivalent containment penetration opening size is not exceeded. This requirement is necessary to ensure that the opening of a containment penetration will not adversely affect the ability of filtered SLCRS to maintain the associated plant area at a negative pressure. SR 4.9.4.4 establishes the maximum equivalent containment penetration opening size for each applicable plant area.

For the purpose of satisfying SR 4.9.4.1, area flow rate is not required to be verified. Each flow path must be verified to be aligned in the correct manner to ensure that the area is being exhausted to at least one OPERABLE filtered SLCRS train. In addition, the term "open containment penetrations" as stated in SR 4.9.4.1 is defined as a penetration that provides direct access from the containment atmosphere to the outside atmosphere. The 12 hour surveillance specified in SR 4.9.4.1.b does not pertain to the containment purge and exhaust containment penetrations provided that containment air is being exhausted through the exhaust penetration to filtered SLCRS. For the purpose of satisfying the requirements of SR 4.9.4.1, it is acceptable for the PAL area to have an observed air flow through the PAL into containment and thereby be considered to be exhausting to filtered SLCRS provided the following conditions have been met: 1) SR 4.9.4.4 has demonstrated that with both air lock doors open and the purge and exhaust containment penetrations closed, the PAL area is maintained negative with respect to atmosphere pressure by the PAL area filtered SLCRS flow path. 2) The PAL area is verified to be serviced (i.e., flow path alignment) by filtered SLCRS from a ventilation flow path other than containment.

INSERT G (Continued)

SR 4.9.4.4 verifies the required plant area(s) integrity and the ability of filtered SLCRS to maintain the area(s) at a negative pressure with open containment penetrations. The ability of filtered SLCRS to maintain a negative pressure in the required plant area(s) provides assurance that radioactivity that may be released through open containment penetrations, due to a fuel handling accident occurring inside containment, is collected and filtered for iodine removal prior to discharge to the atmosphere. The negative pressure with respect to atmosphere includes the verification of negative pressure of $\leq$ -0.125 inches of water gauge with respect to adjacent plant areas (excluding containment) that are not being serviced by filtered SLCRS as well as environmental atmosphere pressure. The purge and exhaust containment penetrations need to be isolated during performance of SR 4.9.4.4. The isolation of these containment penetrations is necessary to accurately reflect the plant conditions following a fuel handling accident inside containment. These containment penetrations will be automatically isolated by a high radiation signal from the containment purge exhaust radiation monitors. Therefore, SR 4.9.4.4 can not be performed with this additional SLCRS filtered flow path in service. SR 4.9.4.4 requires that the maximum equivalent containment penetration opening size for each applicable plant area be established. This requirement is necessary to ensure that the opening of containment penetrations will not adversely affect the ability of filtered SLCRS to maintain the associated plant area at a negative pressure. The establishment of the maximum equivalent containment penetration opening size for each applicable plant area involves the measurement of the filtered SLCRS exhaust flow rate and the negative pressure for the applicable plant area. Utilizing this data, a maximum equivalent containment penetration opening size can be calculated. The available margin between the measured area negative pressure and the minimum required area negative pressure is utilized to allow opening of containment penetrations. For the PAL area, the establishment of the maximum equivalent containment penetration opening size is accomplished by performing SR 4.9.4.4 with both doors of the PAL open. If the PAL is the only containment penetration in the PAL area that will be opened, then a calculation of the maximum equivalent containment penetration opening size is not required. The performance of SR 4.9.4.4 with both doors of the PAL open establishes the maximum equivalent containment penetration opening size for the PAL area. The area where the open containment penetration is located may be defined as containing more than one room. For example, if two rooms are connected via an open doorway, the area can be defined as both rooms provided that this area is being exhausted to filtered SLCRS. All doors to this area are required to be closed except as noted in Footnote (1). This footnote provides an exception to the requirement that all doors to the area are closed to allow for entry and exit. This footnote is not intended to permit doors to be blocked opened.

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

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

← 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. ~~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.~~

REPLACE WITH INSERT "H" →

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% 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.

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

INSERT H

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.

ATTACHMENT B

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 155
REVISION OF REQUIREMENTS ASSOCIATED WITH CONTAINMENT CLOSURE

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed amendment will revise Table 3.3-6 titled "Radiation Monitoring Instrumentation," Item 2.c.ii. 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 2.c.ii will be revised from Mode "6" to table notation (5). This new table notation (5) will specify the following: "During movement of fuel assemblies within containment." The Modes in which the surveillance is required as specified in Table 4.3-3 titled "Radiation Monitoring Instrumentation Surveillance Requirements" for Item 2.c.ii will be revised from Mode "6" to Footnote ###. This new footnote states the following: "During movement of fuel assemblies within containment."

Limiting Condition For Operation (LCO) 3.9.4 titled "Containment Building Penetrations" will be revised to allow both doors on the Personnel Air Lock (PAL) to be open provided certain conditions are met. These conditions include the requirement that at least one of the PAL doors is capable of being closed, that a designated individual is available to close at least one PAL door, that the PAL area is being exhausted to at least one operable filtered Supplemental Leak Collection and Release System (SLCRS) train with all doors (except for the air lock doors) to the PAL area closed, and that SR 4.9.4.4 has been satisfied with both PAL doors open. The requirement for the PAL area doors to be closed is modified by a Footnote (1). This footnote provides an exception to this requirement for the purpose of entry and exit.

LCO 3.9.4.c will be modified by adding the words "(excluding the PAL)" following the words "Each penetration." LCO 3.9.4.c will also be revised to allow a containment penetration to be open provided certain conditions are met. These conditions include the requirement that the penetration is capable of being closed by an isolation valve, blind flange, manual valve, or approved functional equivalent; that the maximum equivalent containment penetration opening size for the associated plant area is not exceeded; that a designated individual is available to close the penetration; and that 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. The requirement for the area doors to be closed is also modified by Footnote (1). The current 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 SLCRS train." LCO 3.9.4 will also be revised by removing the terms "Core Alterations or" and "irradiated" from the LCO Applicability and Action requirements. In addition, the term "assemblies" will be added to the LCO Mode Applicability and Action requirements following the word "fuel."

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. The current 7 day surveillance interval will be designated as Part "a" of SR 4.9.4.1. A new surveillance requirement, designated as Part "b" of SR 4.9.4.1, will state the following: "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)]." Footnote (1) will also modify this surveillance requirement. SR 4.9.4.2.a will be modified by revising the words "through the" to the words "to filtered." SR 4.9.4.2.c will be replaced with a new SR 4.9.4.3. The proposed SR 4.9.4.3 will state 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. A new SR 4.9.4.4 will be added. The proposed SR 4.9.4.4 will require that those areas required to be exhausted to filtered SLCRS (except for the containment) be verified at least once per 7 days to be maintained at a negative pressure of ≤ - 0.125 inches of water gauge with respect to atmospheric pressure when operating filtered SLCRS. SR 4.9.4.4 will also require that the maximum equivalent containment penetration opening size for each applicable plant area be established.

LCO 3.9.9 titled "Containment Purge and Exhaust Isolation System" will be revised by removal of terms "Core Alterations or" and "irradiated" from the LCO Applicability requirements. In addition, the term "assemblies" will be added to the LCO Applicability following the word "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."

Editorial and format changes are also included. These changes include revision of Index Pages to reflect changes in Bases page numbering due to the addition of text, updating to current page format, the addition of new technical specification pages to accommodate the addition of text, punctuation changes, shifting the page footer on page 3/4 3-44 and a change of the spelling of the word "airlock." The Bases section has been revised as necessary to reflect the changes to these Specifications. 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.

B. DESIGN BASES

The fuel handling accident (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. Environmental release from the containment is precluded by a design which automatically isolates the containment following detection of radioactivity by redundant containment purge radiation monitors (2HVS-RQ104A & B). 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 virtually all radioactive releases would be contained in the containment building. Based on this design, a radiological dose analysis for a FHA inside of containment was not required to be performed.

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 release from the fuel 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).

A FHA in the fuel building is currently defined as the dropping of one spent fuel assembly onto another fuel assembly in the spent fuel storage area. The FHA is postulated to cause damage to all fuel rods in the dropped assembly (264 rods) plus an additional 50 rods in the struck fuel assembly with subsequent release of all the activity in the fuel rod gap. 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. The bases for the failure of the postulated 314 rods for a FHA in the fuel building is referenced in a Beaver Valley Power Station (BVPS) Unit No. 2 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. Brobks, et al. The BVPS FHA dose calculation states that the assumption of 50 additional rods versus 34 additional rods adds conservatism. The USNRC staff analysis also 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.

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 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 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.

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 containment penetrations, the containment isolation valves, and the piping between the valves associated with the purge air system are Safety Class 2. The remainder of the system is non-nuclear safety (NNS) class. 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.

A Core Alteration is defined in the BVPS Unit No. 2 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.

For BVPS Unit No. 2, 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. These safety classes correspond to the "Group" described in Regulatory Guide 1.26 as follows: Safety Class 1 is equivalent to "Group A," Safety Class 2 to "Group B," Safety Class 3 to "Group C," and NNS to "Group D."

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.

C. JUSTIFICATION

The proposed deletion of the term "Core Alterations" from LCO 3.9.4 and LCO 3.9.9 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 equipment be operable during plant evolutions where a FHA can occur inside containment; i.e., during movement of fuel assemblies within containment. This proposed change is also consistent with NRC approved Technical Specification Traveler Form (TSTF) 51 Revision 2 to NUREG 1431 Revision 1 titled "Standard Technical Specifications - Westinghouse Plants" (ISTS).

The proposed deletion from LCO 3.9.4 and LCO 3.9.9 of the term "irradiated" modifying the word "fuel" is necessary to ensure that requirements of these specifications continue to apply when moving either a non-irradiated or an irradiated fuel assembly within containment. The dropping of a non-irradiated fuel assembly into the reactor core could result in damage to struck fuel assemblies. The damage to the struck fuel assemblies could result in a radiological release if these assemblies have been irradiated. Currently, the term "Core Alterations" in the LCO applicability requires that LCO 3.9.4 and 3.9.9 be applicable during the movement of any fuel within the reacor 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 change of the minimum number of radiation monitors specified in Table 3.3-6 for Item 2.c.ii from "1" to "2" and the change in the Mode applicability from Mode "6" to "During movement of fuel assemblies within containment" aligns the radiation monitor operability requirements with the containment purge isolation requirements specified in LCO 3.9.9 and 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 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 10 CFR Part 100 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. The proposed change in the mode in which the surveillance is required, as specified in Table 4.3-3 for Item 2.c.ii from Mode "6" to "During movement of fuel assemblies within containment" 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.

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 will permit this required testing and other required surveillance and maintenance activities to occur under the administrative controls described in the proposed technical specifications. It is BVPS's objective to perform future refueling outages in under 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.

The proposed Technical Specification (TS) changes will allow both doors of the containment PAL and other containment penetrations to be open during fuel movement provided certain requirements are met. The NRC staff has established the following generic criteria (as stated in the Safety Evaluation Report for Amendment Number 114 for Callaway Unit 1 dated 7/15/96) for the acceptance of proposed amendments that would allow both doors of the containment PAL to be open during fuel movement or Core Alterations:

1. The radiological consequences for a FHA in containment must meet SRP 15.7.4 acceptance criteria without credit for the mitigation effects of the primary containment.

2. Administrative procedures must be established to:

   a. ensure that appropriate personnel are aware of the OPEN status of the containment during Core Alterations and fuel handling,

   b. ensure that an open air lock is capable of rapid closure (i.e., ≤ 30 minutes), with quick disconnect and removal capability for hoses, cables, ramps, and door seal protective covers, and

   c. ensure that an individual is designated and readily available to close the air lock following the evacuation that would occur in the event of an accident.

The proposed changes to LCO 3.9.4 and associated Bases will ensure that the above administrative procedure requirements will be met (except as noted below) when the PAL is open during movement of fuel within the containment. The proposed Bases wording specifies that these requirements must be met in order to demonstrate that a PAL door is capable of being closed. LCO 3.9.4 will also specify similar requirements for open containment penetrations other than the PAL. Bases wording is also being proposed that states the requirements that must be met in order to demonstrate that a containment penetration is capable of being closed. The term "readily" in reference to an individual's availability was not included in the proposed changes. The proposed requirements of LCO 3.9.4 require that areas containing open containment penetrations are exhausting to filtered SLCRS. Therefore, the timeliness of closing the containment penetration is not as significant in terms of radiological doses as it would be if the open containment penetrations were being exhausted to the environment without filtration. The proposed changes to allow open containment penetrations, including the PAL, are generally consistent with those approved to allow an open air lock at Vogtle Unit Nos. 1 and 2 in Amendment Nos. 105 and 83 dated January 29, 1999 (TAC Nos. MA2196 and MA2197).

SRP 15.7.4 Section II.5 states that "An acceptable alternative approach is containment venting through an ESF atmosphere cleanup system or containment isolation during fuel handling operations." The proposed change will allow a containment penetration to be open during movement of fuel assemblies provided that filtered SLCRS is exhausting the area where the open containment penetration is located. This change meets the criteria stated in SRP 15.7.4 since the open containment penetrations will be venting through an Engineered Safety Feature (ESF) atmosphere cleanup system prior to being vented to the environment. SRP 15.7.4 Section III.4 states that if fuel handling operations occur only when the containment is exhausted to the environment via an ESF filter system, the radiological consequences should be calculated giving appropriate credit for this system. The radiological consequences of a FHA inside containment with open containment penetrations were performed utilizing the same filtration capabilities that were assumed for a FHA in the fuel building.

The results of the dose calculation for a FHA inside the containment building are presented in Attachment C. The attached radiological analysis demonstrates that should a FHA occur inside containment, the projected offsite doses will be well within the applicable regulatory limits of 10 CFR 100.11 of 300 rem thyroid and 25 rem whole body, and are less than the more restrictive guidance criteria in the SRP Section 15.7.4 of 75 rem thyroid and 6 rem whole body. Control room operator doses are less than the 10 CFR Part 50 Appendix A GDC 19 limit of 5 rem whole body or its equivalent to any part of the body. The dose calculation for a FHA inside containment is based on conservatively assuming the airborne release to the environment over a two hour period with credit taken for pre-release filtration by SLCRS and for radioactive decay. The effect of radioactive decay is conservatively minimized by assuming a front-loaded exponential release profile.

The number of fuel rods that are utilized in the attached radiological analysis is based on the USNRC staff analysis referenced in NUREG 0800, SRP Section 15.7.4 titled "Radiological Consequences of Fuel Handling Accident." The USNRC staff analysis 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.

For a FHA in the fuel building, the radioactivity released from the fuel pool into the fuel building atmosphere is filtered by SLCRS. 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 certain filtration assumptions contained in the radiological analysis for the fuel building are met. The proposed changes to LCO 3.9.4 ensure that the SLCRS filtration assumptions in the radiological analysis contained in Attachment C are met. For the PAL doors to remain open during fuel movement inside containment, the PAL area is required to be exhausting to at least one operable filtered SLCRS train. For other containment penetrations (excluding the containment purge and exhaust penetrations) the area outside of containment where the open containment penetration piping is located must be exhausting to at least one operable filtered SLCRS train. In addition, the required area doors must be closed.

The proposed change to LCO 3.9.4 will require that at least 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. This requirement is consistent with the current requirements stated in LCO 3.9.12 and the required ISTS actions for an inoperable fuel building air cleanup system filter train contained in LCO 3.7.13.

The proposed addition of LCO 3.9.4.b.4, which requires that SR 4.9.4.4 has been satisfied with both PAL doors open, is necessary to ensure that the opening of the PAL will not adversely affect the ability of filtered SLCRS to maintain the PAL area at a negative pressure. The proposed exception to LCO 3.9.4.c for the PAL is necessary since specific requirements for the PAL are stated in LCO 3.9.4.b. The proposed wording of LCO 3.9.4.c.1.b requires that a containment penetration may be opened if the maximum equivalent containment penetration opening size for the associated plant area is not exceeded. This requirement is necessary to ensure that the opening of containment penetrations will not adversely affect the ability of filtered SLCRS to maintain the associated plant area at a negative pressure. SR 4.9.4.4 establishes the maximum equivalent containment penetration opening size for each applicable plant area. Proposed Footnote (1) provides a necessary exception to this requirement to allow for personnel entry and exit.

The proposed change to LCO 3.9.4.c.2 will make the LCO terminology more consistent with the proposed Specification 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 .

Based on the proposed changes to LCO 3.9.4 and associated surveillance requirements, the current requirements to maintain the fuel building in a specified condition (i.e., fuel building portion of SLCRS is operating and discharging to at least one train of SLCRS) will be consistent with the conditions for an open containment penetration (excluding the purge and exhaust penetrations). SR 4.9.12 for the fuel building requires the verification that the fuel building portion of SLCRS is operating with all building doors closed at least once per 12 hours. The proposed wording of SR 4.9.4.1.b is based on this fuel building surveillance requirement. SR 4.9.4.1.b requires that at least once per 12 hours that the required areas are verified to be exhausting to filtered SLCRS and that the plant area doors are closed. In the case of the open containment penetrations and the fuel building, the same filtered SLCRS (except for different branch lines to the filter banks) will be relied upon to mitigate a FHA. The filtered portion of SLCRS is classified as QA Category I, Safety Class 3 and Seismic Category I as stated in UFSAR Section 6.5.3.2.1 titled "Design Bases." As described in UFSAR Section 6.5.1 titled "Engineered Safety Feature Filter Systems," filtered SLCRS is considered to be an ESF filter system used to mitigate the consequences of accidents.

The proposed revision of the words "through the" to the words "to filtered" 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 is serviced by SLCRS.

The proposed wording of SR 4.9.4.3 will require that the SLCRS be demonstrated operable per Specification 4.7.8.1 with exception to Item 4.7.8.1.c.2. This new SR incorporates the requirements of the current SR 4.9.4.2.c. The incorporation of SR 4.9.4.2.c into the proposed SR 4.9.4.3 is necessary since SLCRS will be required to be operable for other reasons than just containment purge and exhaust operability.

The proposed addition of SR 4.9.4.4 is necessary in order to periodically verify the integrity of plant areas containing open containment penetrations. This proposed surveillance also establishes the maximum equivalent containment penetration opening size for each applicable plant area. This requirement is necessary to ensure that the opening of a containment penetration will not adversely affect the ability of filtered SLCRS to maintain the associated plant area at a negative pressure. This proposed surveillance requirement is also necessary to ensure that radioactivity that may be released through open containment penetrations following a FHA in containment is filtered prior to being released to the environment. The proposed surveillance interval of at least once per 7 days is consistent with the surveillance interval currently stated in SR 4.9.4.1. In addition, the proposed 7 day surveillance interval is more restrictive than the 18 month surveillance interval stated in ISTS SR 3.7.13.4 for the Fuel Building Air Cleanup System. The required negative pressure of –0.125 inches water gauge is also consistent with the negative pressure requirement stated in ISTS SR 3.7.13.4.

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 the 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.

Testing of the SLCRS ventilation will be required to meet the proposed LCO 3.9.4 and associated SRs. Ventilation flow rate adjustments and/or system modifications may be required in order to meet the proposed requirements.

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. The proposed amendment increases the dose at the site boundary and the control room operator dose due to a fuel handling accident (FHA) occurring within containment; however, the dose remains within acceptable limits. Based on the radiological analysis contained in Attachment C for a FHA inside containment with open containment penetrations being filtered by SLCRS, the resultant radiological consequences of this event will be well within the applicable regulatory limits of 10 CFR 100.11 of 300 rem thyroid and 25 rem whole body, and are less than the more restrictive guidance criteria in the SRP Section 15.7.4 of 75 rem thyroid and 6 rem whole body. Control room operator doses are less than the 10 CFR 50 Appendix A GDC 19 limit of 5 rem whole body or its equivalent to any part of the body. The radiological analysis is based on all airborne activity reaching the containment atmosphere, as a result of a FHA inside containment, being released to the environment over a 2 hour period. The 2 hour release period is based on the guidance contained in Regulatory Guide 1.25 titled "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." The proposed amendment contains the requirement to maintain the capability to close open containment penetrations within 30 minutes following a FHA inside containment. Completion of this action will reduce the dose consequence of a FHA within containment by terminating the release to the environment prior to all airborne activity being released from the containment.

LCO 3.9.10 titled "Water Level – Reactor Vessel" will continue to ensure that at least 23 feet of water is maintained over the fuel during fuel movement when the plant is in Mode 6. 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 150 hours after shutdown. These LCOs will continue to ensure that two of the key assumptions used in the radiological safety analysis 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 to remove the terms "Core Alterations" and "irradiated" from LCO 3.9.4 and LCO 3.9.9 will continue to ensure that plant equipment is available during plant evolutions when a FHA can occur inside containment; i.e., during movement of fuel assemblies within containment.

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 these Core Alteration events.

The proposed wording of LCO 3.9.4 and associated surveillance requirements required that certain plant area doors be closed. Therefore, the exception to this requirement, as provided in proposed Footnote (1), 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 plant area occurs due to the area being at a negative pressure.

The proposed addition of exception to LCO 3.9.4.c for the PAL is administrative in nature and does not affect plant safety.

The proposed change to LCO 3.9.4.c.2, to make the terminology more consistent with the proposed Specification terminology, is administrative in nature and does not affect plant safety.

The proposed revision of the words "through the" to the words "to filtered" in SR 4.9.4.2.a does not change the LCO 3.9.4 requirements. This change makes the LCO and surveillance requirements consistent. This change is administrative in nature and does not affect plant safety.

The proposed revision 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 proposed changes to SR 4.9.4.1 and SR 4.9.9, to remove unnecessary detail on when these surveillances are required to be performed, are administrative in nature and do not affect plant safety.

The editorial and format changes which includes revision of Index pages to reflect changes in Bases page numbers due to the addition of text, updating to current page format, the addition of new technical specification pages to accommodate the addition of text, punctuation changes, shifting of a page footer, a change of the spelling of the word "airlock", and the addition of the term "assemblies" following the word "fuel" are editorial in nature and do not affect plant safety. The Bases section has been revised as necessary to reflect the changes to these Specifications. 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.

Therefore, based on the above, this proposed amendment is considered safe.

E. NO SIGNIFICANT HAZARDS EVALUATION

The proposed amendment will revise the minimum number of channels required to be operable and the Modes in which surveillances are required for the Containment Purge Exhaust Radiation Monitoring Instrumentation.

The proposed amendment will revise the Limiting Condition For Operation (LCO) 3.9.4 titled "Containment Building Penetrations." The proposed amendment will allow both doors on the personnel air lock (PAL) and other containment penetrations to be open during movement of fuel assemblies within containment provided certain conditions are met. This specification will also be revised by removing the terms "Core Alterations" and "irradiated" from the mode applicability and action requirements. Additional changes to the associated surveillance requirements also include the removal of unnecessary detail from Surveillance Requirement (SR) 4.9.4.1, the addition of new surveillance requirements, and the modification of the wording in existing surveillance requirements.

The requirements for the Containment Purge and Exhaust Isolation System contained in Specification 3/4.9.9 will be revised by removing the terms "Core Alterations" and "irradiated" from the specification Applicability requirements.

The associated 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."

The proposed amendment also includes administrative, editorial and format changes. These changes include revision of Index pages to reflect changes in Bases page numbers due to the addition of text, updating to current page format, the addition of new technical specification pages to accommodate the addition of text, punctuation changes, shifting of a page footer, a change of the spelling of the word "airlock", the addition of the words "(excluding the PAL)", a revision to Specification terminology, and the revision of the words "through the" to words "to filtered". In addition, the term "assemblies" will be added following the word "fuel". The Bases section has been revised as necessary to reflect the changes to these Specifications. 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.

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.

   Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   Based on the current technical specification requirements, an environmental release due to a fuel handling accident (FHA) occurring within containment is precluded by a design which automatically isolates the containment following detection of radioactivity by redundant containment purge monitors. The proposed amendment, which permits containment penetrations to be open during movement of fuel assemblies within containment, increases the dose at the site boundary and the control room operator dose due to a FHA occurring within containment; however, the dose remains within acceptable limits. Based on a radiological analysis of a FHA within containment with open containment penetrations being filtered by the Supplemental Leak Collection and Release System (SLCRS), the resultant radiological consequences of this event are well within the 10 CFR Part 100.11 limits, as defined by acceptance criteria in the Standard Review Plan (SRP) Section 15.7.4. Control room operator doses remain less than the 10 CFR Part 50 Appendix A General Design Criteria (GDC) 19 limit of 5 rem whole body or its equivalent to any part of the body. The proposed changes to LCO 3.9.4 and associated surveillance requirements will ensure that SLCRS filtration assumptions in the associated radiological analysis are met.

   LCO 3.9.10 titled "Water Level – Reactor Vessel" will continue to ensure that at least 23 feet of water is maintained over the fuel during fuel movement when the plant is in Mode 6. 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 150 hours after shutdown. 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 revision 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 inside containment.

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. 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.

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 does not impact Technical Specification requirements for systems needed to prevent or mitigate other Core Alteration events. The filtered SLCRS that will be utilized to control and filter the radioactive release from a FHA occurring within containment is the same system (with the exception of the flow path to the filter banks) currently relied upon to control and filter the release from a FHA in the fuel building. 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. This system will be relied upon to control the releases from open containment penetrations should a FHA occur inside of containment until such time that these open containment penetrations can be isolated. The proposed amendment contains the requirement to maintain the capability to close open containment penetrations within 30 minutes following a FHA inside containment.

The filtered SLCRS that will be relied upon to mitigate a FHA within containment is classified as Quality Assurance (QA) Category I, Safety Class 3 and Seismic Category I as stated in Updated Final Safety Analysis Report (UFSAR) Section 6.5.3.2.1 titled "Design Bases." As described in UFSAR Section 6.5.1 titled "Engineered Safety Feature Filter Systems," filtered SLCRS is considered to be an engineered safety features (ESF) filter system used to mitigate the consequences of accidents.

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?

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. The proposed amendment increases the dose at the site boundary and the control room operator dose due to a FHA occurring within containment; however, the dose remains within acceptable limits. The margin of safety as defined by 10 CFR Part 100 has not been significantly reduced.

The revised radiological analysis based on the proposed amendment demonstrates that during a FHA inside containment, the projected offsite doses will be well within the applicable regulatory limits of 10 CFR Part 100.11 of 300 rem thyroid and 25 rem whole body, and are less than the more restrictive guidance criteria in the SRP Section 15.7.4 of 75 rem thyroid and 6 rem whole body. Control room operator doses are less than the 10 CFR Part 50 Appendix A GDC 19 limit of 5 rem whole body or its equivalent to any part of the body. This radiological analysis is based on all airborne activity reaching the containment atmosphere, as a result of a FHA inside containment, being released to the environment over a 2 hour period. The 2 hour release period is based on the guidance contained in Regulatory Guide 1.25 titled "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." The proposed amendment contains a Bases requirement to maintain the capability to close open containment penetrations within 30 minutes following a FHA inside containment. Completion of this action will reduce the dose consequence of a FHA within containment by terminating the release to the environment prior to all airborne activity being released from the containment.

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 revision 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 proposed changes to SR 4.9.4.1 and SR 4.9.9, to remove unnecessary detail on when these surveillances are required to be performed, are administrative in nature and do not affect plant safety.

The proposed revision of the words "through the" to the words "to filtered" in SR 4.9.4.2.a does not change the LCO 3.9.4 requirements. This change makes the LCO and surveillance requirements consistent. This change is administrative in nature and does not affect plant safety.

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. 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.

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 15.7.4 of 75 rem thyroid and 6 rem whole body. Control room operator doses are less than the 10 CFR Part 50 Appendix A GDC 19 limit of 5 rem whole body or its equivalent to any part of the body.

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 D.

ATTACHMENT C

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 155

Safety Analysis of the Radiological Consequences of a Fuel Handling DBA at BVPS Unit 2, Control Room, EAB and LPZ Doses

Duquesne Light
Radiological Controls

REVISION	[redacted]	2	3

Subject
UNIT 2 FUEL HANDLING ACCIDENT DOSES AT EAB, LPZ, AND COMMON CONTROL ROOM

ERS-SFL-89-019

PAGE 1 OF ~~24~~ ~~23~~ 21

Reference
RCM RP ______ RIP   EPP ______   T/S ______   EM ______   DCP ______   Other U2 T/S amend #12

Review Category
[x] RSC Req'd   [ ] RSC Not Req'd

10 CFR 50.59 [ ] Required

Purpose The purpose of this calculation package is to document the re-analysis of the consequences of a Unit 2 UFSAR fuel handling accident with regard to offsite dose consequences and common control room design habitability. This re-analysis was requested by DLC Licensing in response to the safety evaluation report prepared by the USNRC on the Unit 2 T/S amendment 12, in which the fuel gap fraction for I-131 was stated as 12% in lieu of the 10% previously stated by Regulatory Guide 1.25.

The change in the postulated common control room habitability with delayed isolation is addressed in calculation ERS-SFL-89-007.

Rev.	Signatures	Description
	BY Date CHK Date APP Date	
3	BY Date CHK Date APP Date	
2	BY 3-20-00 Date CHK 3/20/00 Date APP FOR THE RSC 5-00 3/23/00 Date	Assumes maximum of 617 fuel rods breached, considering that the accident could occur within containment with penetrations open. Makes 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.
1	BY 5-28-99 Date CHK 5-28-99 Date APP for RSC 09-99 6-3-99 Date	Changed core inventory to one calculated using ORIGENS and limiting fuel enrichment parameters. Used current, approved atmospheric dispersion factors. Changed release duration for control room dose analysis consistent with UFSAR & offsite dose analysis.
0	BY 7/4/89 Date CHK 7/7/89 Date APP FOR RSC 15-89 7/28/89 Date	See Checklist / Attachments

Checklist
- [x] PURPOSE
- [x] ASSUMPTIONS
- [x] METHODOLOGY
- [x] INPUT DATA
- [x] RESULTS
- [x] REFERENCES

Attachments
- [x] DATA SHEETS
- [ ] ILLUSTRATIONS
- [x] PRINTOUTS
- [ ] CODE LISTING

■ Document Control
■ Calculation File
[ ] MGR, RADCON
[x] DIR, RadEng
[ ] DIR, RadOps-1
[ ] DIR, RadOps-2
[ ] DIR, RadHealth
[ ] DIR, Eff. & Env.
■ Trng. Dept.
■ Author: S.F. LaVie
[x] ~~J.V. Vassello~~ 6-26-99 JM
[x] R W Fedor NS+L

RE1.103-1   6/88

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) occurring within the reactor containment building at Beaver Valley Unit 2. Doses for the whole body (effective dose equivalent - EDE), the thyroid (committed dose equivalent - CDE) and the skin (skin dose equivalent - SDE) are calculated. The plant parameters and assumptions used herein are consistent with the plant design basis, 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 prompted by management direction to pursue removal of certain aspects of the containment closure requirement during periods of core alteration. Specifically, the analysis is performed to demonstrate that the accident control room operator and offsite doses will be within applicable limits and criteria if the accident occurs when the personnel airlock and/or penetrations to contiguous areas are open. (This calculation does not address releases from the containment building equipment hatch or via the Primary Auxiliary Building ventilation system. These release pathways are unfiltered and have different atmospheric dispersion factors such that this calculation does not bound the accident doses where these pathways would be used.) Currently, Technical Specification 3/4.9.4[1] specifies closure "during core alterations or movement of irradiated fuel within the containment". With this requirement, previous evaluations have credited radiation monitor initiated ventilation isolation following an FHA, effectively preventing an uncontrolled release from occurring. Consequently, there is no current design basis analysis for an FHA within containment documented in the UFSAR. This analysis is intended to support a license amendment to modify the Technical Specification containment closure requirement so that core alterations or fuel movement may be conducted with either the personnel airlock or other penetrations being open. The nature of this analysis is such that it bounds the radiological consequences of an FHA within the fuel building. The release source term for the accident in containment is higher (617 rods broken versus 314 in the fuel building), and all other accident conditions and assumptions are identical.

Fuel Handling Accident

This DBA is described in the Unit 2 UFSAR[2] and NUREG 0800 Chapter 15, Section 15.7.4[3]. The accident occurs while moving a fuel assembly from the reactor vessel to fuel pool storage. The assembly is dropped, resulting in rupture of all fuel rods and release of radioactive iodine and noble gas into the pool water. The dropped assembly strikes other assemblies additional damage. The maximum number of rods breached as a consequence of a drop over the core (617 rods) is taken from the same reference document[4] that provides the value for the accident occurring in the fuel building (314 rods). This is discussed in more detail in the Input Data and Assumptions section of this calculation. All noble and gas radioactivity in the rod gaps is assumed to be immediately released to the fuel pool.

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 assemblies by a factor of 100 before it becomes airborne within the building. After becoming airborne, it is conservatively assumed to be released to the environment over a two hour period with credit taken for pre-release filtration by the SLCRS charcoal and HEPA filters, and for radioactive decay. The effect of radioactive decay is conservatively minimized by assuming a front-loaded, exponential release profile. Additional details are provided below.

The Control Room is expected to be isolated and pressurized with bottled air (60 minute supply) and filtered forced ventilation (after 60 minutes). This will be initiated within a short period following the accident by either the control room radiation monitor alarm and associated automatic action, or by the operators following the Abnormal Operating Procedure[5] which provides response directions specific to the FHA. This analysis conservatively takes no credit for these actions The Control Room ventilation system is assumed to remain in the normal, unfiltered configuration for the duration of the release.

In addition to the above multiple, conservative assumptions, administrative controls may be established to close containment within a reasonable time following the accident. This should be accomplished prior to the end of the activity release, further reducing the accident consequences. This action is not credited in this analysis. Likewise, no such action is assumed should the accident occur in the fuel building. Further discussion of mitigating administrative controls that may be established is outside the scope of this calculation.

Calculation History

<u>SWEC 12241 UR(B)-189-0 (1982)</u>[6]
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.

<u>SWEC 12241 UR(B)-280-0 (1983)</u>[7]
Initial calculation of Control Room operator dose. Superseded by SWEC 12241 UR(B)-295-0.

<u>SWEC 12241 UR(B)-189-1 (1984)</u>[8]
Revision to update atmospheric dispersion factor for the EAB. Superseded by ERS-SFL-89-019.

<u>SWEC 12241 UR(B)-295-0 (1984)</u>[9]
Calculated control room operator dose, with different assumptions for operator action to actuate the control room emergency ventilation system.

<u>SWEC 12241 UR(B)-445-0 (1987)</u>[10]
Calculated control room operator dose for the combined (Unit 1 + Unit 2) control room. Superseded by ERS-SFL-89-019-0.

<u>ERS-SFL-89-007-0 (1989)</u>[11]
Special calculation to assess effect of delaying control room isolation for 10 minutes following an FHA.

<u>ERS-SFL-89-019-0 (1989)</u>[12]
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 *****

<u>ERS-SFL-93-004-0 (1993)</u>[13]
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)[14]
Used revised reactor core radionuclide inventory calculated using updated fuel parameters (high burn-up) and the computer code ORIGENS[15]; 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[16,17]. This calculation is intended to replace ERS-SFL-89-019[12] and ERS-SFL-93-004[13], but at the time of this writing has not been submitted for NRC review.

This Calculation, ERS-SFL-89-019-2 (2000)
Assumes maximum of 617 fuel rods breached. Makes 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.

METHODOLOGY

Overall Methodology

The methodology used in this analysis is similar to that used in previous analyses. This revision uses version 1.0a 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. This calculation is the first use of version 1.0a, which performs the calculations identically to version 1.0.

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 FHA is based on the guidance provided in NUREG-0800, Chapter 15.7.4[3] and USNRC Safety Guide 1.25[19]. One exception to these is that the gap fraction for I-131 is increased from 10% to 12%[20,21,22].

1.2 A minimum 100 hours between reactor shutdown from subcriticality to the accident release.

Although the minimum time is established by Technical Specification[23] as 150 hours, 100 hours was used in prior analyses[6,7,8,9,10,11,12,13,14] and is described in the UFSAR Section 15.7.4.3[2].

1.3 Radioactivity release to the building is assumed to occur instantaneously, and then to the environment over a two hour period.

This approach is conservative particularly for the EAB 0-2 hour dose and is consistent with UFSAR Section 15.7.4.3[2]. Some[6,8,12,14] of the prior analyses assumed an instantaneous release to the environment, while others[7,9,10,11,13] assumed a release rate constant equal to the release flow rate divided by the building volume. 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.

1.4 The maximum number of ruptured fuel rods = 617 rods[4] (containment drop)
314 rods (fuel building drop)

This value is from an analysis performed and docketed by the NRC staff in response to a question posed by the ASLB associated with licensing of a similarly designed facility. Quoting that analysis:

"We believe that the analysis performed was conservative in that it tended to maximize the amount of fuel rods that could be damaged. Key assemblies of conservatism in this analysis were as follows:

(1) The maximum kinetic energy calculated for a dropped fuel assembly was used for the analysis.

(2) All of the kinetic energy was assumed to break fuel rods. No allowance was made for deformation and energy absorption to the top and bottom structure of the fuel assembly, the side retaining structure of the assembly, the spent fuel rack structure, or deformation or energy absorption in the fuel pellets.

(3) A spent fuel assembly was always assumed to drop onto one or more other fuel assemblies when possible (in the core or the spent fuel pool). In the canal or transfer tube, since the assembly is removed from others, damage to only one assembly can result.

(4) No allowance was taken for any buoyancy effects in the water. This effect would reduce the effective weight and hence the kinetic energy dissipation of the fuel assembly falling through the water.

(5) No allowance was taken for the hydrodynamic drag and energy dissipation of the fuel assembly falling through the water.

We believe these to be significant conservatisms that lead to a substantial over-estimate of the amount of damaged fuel that could result from a dropped fuel assembly."

Based on the above, 617 damaged rods is judged to be sufficient and conservative for use in this analysis.

Because the containment drop is bounding, a fuel building drop is not analyzed herein.

2.0. Input Data

2.1 Core activity in gap: [19, 20]

I-131 = 12%
Other iodines = 10%
Kr-85 = 30%
Other noble gases = 10%

2.2 Radial peaking factor = 1.65 [19, 24, 25]

2.3 Number of assemblies in core = 157 [24, 25]

2.4 Number of rods in assembly = 264 [24, 25]

2.5 Pool iodine DF = 100 [19, 25]

2.6 Core Inventory at T = 100 hours, and release to the building: [26, 27]

Refer to Attachment 2 for the activity release to the building calculation.

2.7 Control room air intake and exhaust flow rate = 500 cfm [24, 25, 28]

The control room ventilation system is maintained in the normal configuration for release duration. No credit is taken for isolation or filtration.

2.8 Control room volume = 1.73E+05 ft^3 [24, 25, 28]

2.9 Control room occupancy factors: [29]

0 to 24 hours = 1.0
24 hours to 4 days = 0.6
4 to 30 days = 0.4

2.10 Control room operator breathing rate = 3.47E-04 m^3/s [30]

2.11 Offsite breathing rates: [31]

0 to 8 hours = 3.47E-04 m^3/s
8 to 24 hours = 1.75E-04 m^3/s
24 hours to 30 days = 2.32E-04 m^3/s

2.12 Atmospheric dispersion factors: [32,33,34,35,36]

	EAB (s/m^3)	LPZ (s/m^3)	Control room (s/m^3)
0 – 2 hours	1.25E-03		
0 – 8 hours		6.04E-05	1.20E-04 SLCRS

This analysis assumes that all post-accident leakage from the containment is collected and filtered prior to release via the SLCRS. This assumption is consistent with the current design basis for the accident occurring in the fuel building.

2.13 Building-to-environment release rate constant (λ) = 1.91882E-03 s^{-1}

99.9999% of the radioactivity is assumed to be released over a two hour period.

$$0.000001 = e^{-\lambda t}$$
$$\ln(0.000001) = \lambda t$$
$$13.8155/t = \lambda$$
where $t = 7200$ *seconds,* $\lambda = 1.91882\text{E-}03\ s^{-1}$

2.14 SLCRS main filter banks iodine removal efficiency (total) = 95% [12,13,37,38]

RESULTS AND CONCLUSIONS

The data given above were used to build the TRAILS_PC input files and these were run. The input and output files are given in Attachment 3. The control room operator, EAB and LPZ calculated doses are summarized below.

Offsite doses are well within the applicable regulatory limits of 10 CFR 100.[7]11[39] of 300 rem thyroid and 25 rem whole body, and are less than the more restrictive guidance criteria in the Standard Review Plan[3] of 75 rem thyroid and 6 rem whole body. Control room operator doses are less than the 10 CFR 50[40] Appendix A GDC 19 limit of 5 rem whole body or its equivalent to any part of the body.

	Thyroid CDE (rem)	EDE (rem)	Skin DE (rem)
Control room	3.46E+00	8.51E-03	3.78E-01
EAB	3.72E+01	1.86E+00	4.09E+00
LPZ	1.80E+00	8.96E-02	1.98E-01

REFERENCES

1. BVPS Unit 2 Technical Specification 3/4.9.4, Containment Building Penetrations
2. BVPS Unit 2 UFSAR Chapter 15, Section 15.7.4, Fuel Handling Accident
3. NUREG 0800, USNRC Standard Review Plan, Chapter 15, Section 15.7.4, Radiological Consequences of Fuel Handling Accidents, Rev. 1 July 1981
4. Westinghouse letter DMW-4707, December 6, 1982 and attached Supplemental Testimony, Docket No. STN 50-516/50-517.
5. BVPS Operations Manual Procedure 2OM-53C.4.2.49.1 Irradiated Fuel Damage While Refueling
6. SWEC Calculation Package 12241 UR(B)-189-0, FSAR Section 15.7.4 - Fuel Handling Accident: Releases and Doses, 1982
7. SWEC Calculation Package 12241 UR(B)-280-0, Control Room Dose Due to A Fuel Handling Accident - Normal Operational and Accident Ventilation Systems, 1983
8. SWEC Calculation Package 12241 UR(B)-189-1, FSAR Section 15.7.4 - Fuel Handling Accident: Releases and Doses, 1984
9. SWEC Calculation Package 12241 UR(B)-295, Control Room Doses Due to A Fuel Handling Accident as A Function of Operator Response Time, 1984
10. SWEC Calculation Package 12241 UR(B)-445-0, Control Room Habitability Due to Design Basis Accidents (Except LOCA) at BV2, 1987
11. DLCo Calculation Package ERS-SFL-89-007-0, Combined Control Room Doses Due to DBAs at Unit 2 With Delayed Isolation, 1989
12. DLCo Calculation Package ERS-SFL-89-019-0, Unit 2 Fuel Handling Accident Doses at EAB, LPZ, and Common Control Room, 1989
13. 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
14. DLCo Calculation Package ERS-SFL-89-019-1, Unit 2 Fuel Handling Accident Doses at EAB, LPZ, and Common Control Room, 1999
15. ORIGENS Computer Code, Radiation Safety Information Computational Center, U. S. Department of Energy, Oak Ridge TN (as used by Stone & Webster Engineering Corporation)
16. 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
17. DOE/EH-0070, U. S. Department of Energy, External Dose-Rate Conversion Factors for Calculation of Dose to the Public, July 1988
18. DLCo Calculation Package ERS-SFL-96-004-1, TRAILS_PC: Transport of radioactive Material in Linear Systems, PC version v1.0a, 2000

REFERENCES (cont)

19. USNRC Safety Guide 25 (Regulatory Guide 1.25), Assumptions used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility of Boiling and Pressurized Water Reactors, 1972

20. USNRC NUREG/CR-5009, Assessment of the Use of Extended Burn-up Fuel in Light Water Power Reactors, 1988

21. USNRC Safety Evaluation Report, Beaver Valley Power Station Unit No. 2, Chapter 15 Section 15.7.4

22. USNRC Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 12 to Facility Operating License No, NPF-73, Beaver Valley Power Station Unit No. 2, Docket No. 50-412, 1989

23. BVPS Unit 2 Technical Specification 3/4.9.3, Decay Time

24. DLCo NED Response to EM #116251, Design Basis Accident Parameter Verification - Unit 2, December 4, 1998

25. 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

26. 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

27. SWEC Calculation Package 12241 UR(B)-482-0, Gamma Ray Source Strength for an Irradiated Fuel Assembly as a Function of Time After Reactor Shutdown, 1999

28. BVPS Unit 2 UFSAR Chapter 6, Table 6.4-1, Control Room Envelope Ventilation Design Parameters

29. USNRC NUREG-0800, Standard Review Plan, Chapter 6, Section 6.4, Control Room Habitability System, Revision 2, July 1981

30. Murphy, K. G. and K. M. Campe, Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19, 13th AEC Air Cleaning Conference, 1975

31. USNRC Regulatory Guide 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, 1974

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 2 UFSAR Chapter 15, Table 15.0-14, Control Room Accident Meteorological Parameters

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

36. BVPS Unit 2 UFSAR Chapter 15, Table 15.0-11, Accident Meteorological Parameters

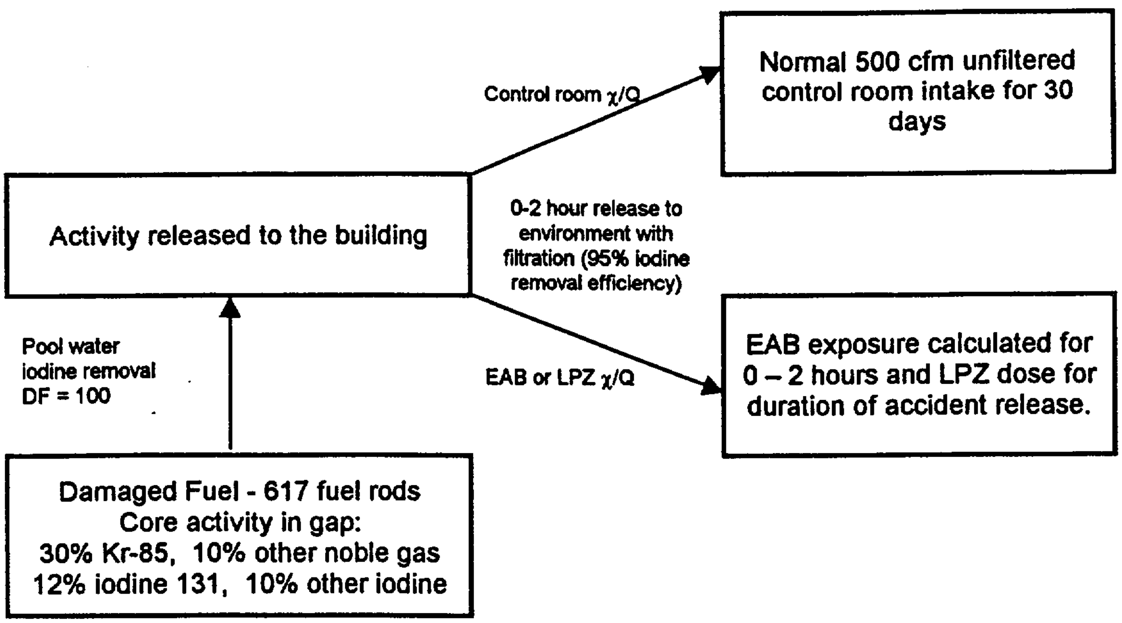
37 BVPS Unit 2 Technical Specification 3/4.7.8 Supplemental Leak Collection and Release System (SLCRS)

38. USNRC Regulatory Guide 1.52, Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants, Revision 2, 1978

39. Title 10 Code of Federal Regulations, Part 100 §100.11, Determination of exclusion area, low population zone, and population center distance

40. Title 10 Code of Federal Regulations, Part 50, Appendix A, Criterion 19, Control Room

ATTACHMENT 1
Fuel Handling Accident Release Model



Fuel Handling Accident Release Source Term Determination

	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	617 Rods FB Activity (µCi)
			=Col B / Col C		=MAX(Col D:Col E)		=Col F * Col G		=Col H * Col I			=Col H * Col I * (J5/264) (J5/264) * 1E+06
Kr-85m	3.64E+00	157	2.32E-02	1.93E-02	**2.32E-02**	1.65	3.83E-02	0.10	3.83E-03	1.00	617	8.94E+03
Kr-85	7.59E+05	157	4.83E+03	6.60E+03	**6.60E+03**	1.65	1.09E+04	0.30	3.27E+03	1.00	617	7.64E+09
Xe-131m	9.19E+05	157	5.85E+03	6.41E+03	**6.41E+03**	1.65	1.06E+04	0.10	1.06E+03	1.00	617	2.47E+09
Xe-133m	1.92E+06	157	1.22E+04	1.22E+04	**1.22E+04**	1.65	2.02E+04	0.10	2.02E+03	1.00	617	4.72E+09
Xe-133	1.03E+08	157	6.56E+05	6.47E+05	**6.56E+05**	1.65	1.08E+06	0.10	1.08E+05	1.00	617	2.53E+11
Xe-135m	6.04E+02	157	3.85E+00	3.82E+00	**3.85E+00**	1.65	6.35E+00	0.10	6.35E-01	1.00	617	1.48E+06
Xe-135	1.99E+05	157	1.27E+03	1.25E+03	**1.27E+03**	1.65	2.09E+03	0.10	2.09E+02	1.00	617	4.89E+08
I-131	5.23E+07	157	3.33E+05	3.38E+05	**3.38E+05**	1.65	5.58E+05	0.12	6.69E+04	0.01	617	1.56E+09
I-132	4.41E+07	157	2.81E+05	2.81E+05	**2.81E+05**	1.65	4.64E+05	0.10	4.64E+04	0.01	617	1.08E+09
I-133	5.45E+06	157	3.47E+04	3.42E+04	**3.47E+04**	1.65	5.73E+04	0.10	5.73E+03	0.01	617	1.34E+08
I-135	3.71E+03	157	2.36E+01	2.34E+01	**2.36E+01**	1.65	3.90E+01	0.10	3.90E+00	0.01	617	9.11E+04

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.

[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.

[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.

TRAILS_PC Input File for Control Room / LPZ (0-30 day) and (0-2 hour) EAB dose

```
'L1 ',6,1.0E-3,1.0E-4,1.0E-5,1.0E-4,1.0,1.0E-4,3.47E-4,1,24,1,2
'FHA - CR, EAB, & LPZ 100 h decay, 617 rods, SLCRS filt, no CR isol, CST
[2x617.in]'
'C1 ','N/A ',0.0,0.0,0.0,0.0
'C2 ','CNMT',1.91882E-3,0.0,0.95,0.0
'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 ',0.0,8.94E3,7.64E9,4*0.0,2.47E9,4.72E9,2.53E11,
1.48E6,4.89E8,0.0,0.0,1.56E9,1.08E9,1.34E8,0.0,9.11E4,5*0.0
'INI',1.0,' ',24*0.0
'TIM',7200.,14400.,28800.,86400.,3.456E5,2.592E6
'XPR',6*0.0
'XPR',6*0.0
'XPR',6*50.0
'XRM',6*0.0
'XRM',1.0,5*0.0
'XRM',6*50.0
'XRF',6*0.0
'XRF',1.0,5*0.0
'XRF',6*0.0
'XOQEB',1.25,5*0.0
'XBREB',3.47,5*0.0
'XOQLZ',6.04,5*0.0
'XBRLZ',3.47,5*0.0
'XOQ',1.20,5*0.0
'XBR',6*1.0
'OCC',4*1.0,0.6,0.4
```

This analysis is performed using Unit 2 bounding parameters, and assumptions from current NRC guidance for source term use in DBAs [NUREG-0800& SG 24].

+ Current dose quantities are calculated.
+ The release rate constant corresponds to 99.9999% of the activity in the fuel building being released in 0-2 hours.
+ The activity release is for the number of rods listed in the header from a maximum activity assembly.
+ Credit is taken for continuous SLCRS activity collection and filtration with 95% iodine removal efficiency. 100% activity collection/filtration is assumed.
+ Control room ventilation configuration - Normal configuration is maintained for the accident duration.

TRAILS_PC -- Transport of Radioactive Material in Linear Systems, V1.0a
FHA - CR, EAB, & LPZ 100 h decay, 617 rods, SLCRS filt, no CR isol, CST [2x617.i ***PROGENY INGROWTH ON ***

	COMP: N/A	COMP: CNMT	COMP: Control Room VOLUME: 1.730E+05 Cu.Ft.
INITIAL:	0.000E+00 Kr-85m	8.940E+03 Kr-85m uCi	0.000E+00 Kr-85m
	0.000E+00 Kr-85	7.640E+09 Kr-85	0.000E+00 Kr-85
	0.000E+00 Xe-131m	2.470E+09 Xe-131m	0.000E+00 Xe-131m
	0.000E+00 Xe-133m	4.720E+09 Xe-133m	0.000E+00 Xe-133m
	0.000E+00 Xe-133	2.530E+11 Xe-133	0.000E+00 Xe-133
	0.000E+00 Xe-135m	1.480E+06 Xe-135m	0.000E+00 Xe-135m
	0.000E+00 Xe-135	4.890E+08 Xe-135	0.000E+00 Xe-135
	0.000E+00 I-131	1.560E+09 I-131	0.000E+00 I-131
	0.000E+00 I-132	1.080E+09 I-132	0.000E+00 I-132
	0.000E+00 I-133	1.340E+08 I-133	0.000E+00 I-133
	0.000E+00 I-135	9.110E+04 I-135	0.000E+00 I-135
ACT MULT (to uCi):	1.000E+00	1.000E+00	1.000E+00

TRAILS_PC -- Transport of Radioactive Material in Linear Systems, v1.0a
FHA - CR, EAB, & LPZ 100 h decay, 617 rods, SLCRS filt, no CR isol, CST [2x617.i ***PROGENY INGROWTH ON ***

REMOVAL:	0.000E+00 1/sec	1.919E-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	9.500E-01	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	7.200E+03	0.00	0.00	0.00	0.00	1.00	1.00	50.0	50.0	0.00
2	1.440E+04	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
3	2.880E+04	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
4	8.640E+04	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
5	3.456E+05	0.00	0.00	0.00	0.00	0.00	0.00	50.0	50.0	0.00
6	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-04	3.470E-04	1.000E+00	1.000E-03	1.000E-04	1.000E-05	1.000E-04
MULTIPLIERS====>								
STEP	TIME,s							
1	7.200E+03	1.20	1.00	1.00	1.25	3.47	6.04	3.47
2	1.440E+04	0.00	1.00	1.00	0.00	0.00	0.00	0.00
3	2.880E+04	0.00	1.00	1.00	0.00	0.00	0.00	0.00
4	8.640E+04	0.00	1.00	1.00	0.00	0.00	0.00	0.00
5	3.456E+05	0.00	1.00	0.600	0.00	0.00	0.00	0.00
6	2.592E+06	0.00	1.00	0.400	0.00	0.00	0.00	0.00

TRAILS_PC -- Transport of Radioactive Material in Linear Systems, v1.0a
FHA - CR, EAB, & LPZ 100 h decay, 617 rods, SLCRS filt, no CR isol, CST [2x617.i ***PROGENY INGROWTH ON ***

STEP	TIME	N/A CURRENT uCi	N/A INTEGRD uCi-sec	CNMT CURRENT uCi	CNMT 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		8.940E+03				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	6.561E-03	4.557E+06	8.744E+03	1.214E+00	1.816E-01	3.706E-11	7.246E+02
2	4.0000 h	0.000E+00	0.000E+00	4.815E-03	4.063E+01	0.000E+00	0.000E+00	9.419E-02	1.923E-11	9.586E+02
3	8.0000 h	0.000E+00	0.000E+00	2.593E-03	5.169E+01	0.000E+00	0.000E+00	2.535E-02	5.175E-12	7.553E+02
4	24.0000 h	0.000E+00	0.000E+00	2.181E-04	5.526E+01	0.000E+00	0.000E+00	1.330E-04	2.715E-14	2.767E+02
5	96.0000 h	0.000E+00	0.000E+00	3.167E-09	5.075E+00	0.000E+00	0.000E+00	7.303E-15	1.491E-24	1.459E+00
6	720.0000 h	0.000E+00	0.000E+00	0.000E+00	7.369E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	8.012E-11
Kr-85m	TOTALS		0.000E+00		4.557E+06	8.744E+03				2.717E+03
Kr-85	INITIAL	0.000E+00		7.640E+09				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	7.640E+03	3.982E+12	7.640E+09	1.061E+06	1.828E+05	3.732E-05	6.961E+08
2	4.0000 h	0.000E+00	0.000E+00	7.640E+03	5.501E+07	0.000E+00	0.000E+00	1.292E+05	2.638E-05	1.112E+09
3	8.0000 h	0.000E+00	0.000E+00	7.640E+03	1.100E+08	0.000E+00	0.000E+00	6.458E+04	1.318E-05	1.342E+09
4	24.0000 h	0.000E+00	0.000E+00	7.639E+03	4.400E+08	0.000E+00	0.000E+00	4.028E+03	8.222E-07	1.257E+09
5	96.0000 h	0.000E+00	0.000E+00	7.635E+03	1.979E+09	0.000E+00	0.000E+00	1.522E-02	3.107E-12	8.362E+07
6	720.0000 h	0.000E+00	0.000E+00	7.600E+03	1.711E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.160E+02
Kr-85	TOTALS		0.000E+00		4.001E+12	7.640E+09				4.491E+09
Xe-131m	INITIAL	0.000E+00		2.470E+09				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	2.458E+03	1.287E+12	2.469E+09	3.429E+05	5.895E+04	1.203E-05	2.246E+08
2	4.0000 h	0.000E+00	0.000E+00	2.446E+03	1.766E+07	0.000E+00	0.000E+00	4.147E+04	8.465E-06	3.578E+08
3	8.0000 h	0.000E+00	0.000E+00	2.423E+03	3.506E+07	0.000E+00	0.000E+00	2.052E+04	4.189E-06	4.288E+08
4	24.0000 h	0.000E+00	0.000E+00	2.330E+03	1.369E+08	0.000E+00	0.000E+00	1.231E+03	2.513E-07	3.949E+08
5	96.0000 h	0.000E+00	0.000E+00	1.957E+03	5.543E+08	0.000E+00	0.000E+00	3.905E-03	7.972E-13	2.520E+07
6	720.0000 h	0.000E+00	0.000E+00	4.300E+02	2.265E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.995E+01
Xe-131m	TOTALS		0.000E+00		1.290E+12	2.469E+09				1.431E+09
Xe-133m	INITIAL	0.000E+00		4.720E+09				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	4.597E+03	2.455E+12	4.711E+09	6.543E+05	1.113E+05	2.273E-05	4.257E+08
2	4.0000 h	0.000E+00	0.000E+00	4.478E+03	3.267E+07	0.000E+00	0.000E+00	7.666E+04	1.565E-05	6.690E+08
3	8.0000 h	0.000E+00	0.000E+00	4.248E+03	6.281E+07	0.000E+00	0.000E+00	3.634E+04	7.419E-06	7.778E+08
4	24.0000 h	0.000E+00	0.000E+00	3.440E+03	2.206E+08	0.000E+00	0.000E+00	1.836E+03	3.747E-07	6.657E+08
5	96.0000 h	0.000E+00	0.000E+00	1.331E+03	5.758E+08	0.000E+00	0.000E+00	2.685E-03	5.482E-13	3.542E+07
6	720.0000 h	0.000E+00	0.000E+00	3.552E-01	3.634E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.181E+01
Xe-133m	TOTALS		0.000E+00		2.456E+12	4.711E+09				2.574E+09

TRAILS_PC -- Transport of Radioactive Material in Linear Systems, v1.0a
FHA - CR, EAB, & LPZ 100 h decay, 617 rods, SLCRS filt, no CR isol, CST [2x617.i ***PROGENY INGROWTH ON ***

STEP	TIME	N/A CURRENT uCi	N/A INTEGRD uCi-sec	CNMT CURRENT uCi	CNMT INTEGRD uCi-sec	RELEASED uCi	AVERAGE RELEASE uCi/sec	CONTROL ROOM CURRENT uCi	CONTROL ROOM CURRENT uCi/cc	CONTROL ROOM INTEGRD uCi-sec
Xe-133	INITIAL	0.000E+00		2.530E+11				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	2.503E+05	1.317E+14	2.528E+11	3.511E+07	6.018E+06	1.229E-03	2.295E+10
2	4.0000 h	0.000E+00	0.000E+00	2.476E+05	1.792E+09	0.000E+00	0.000E+00	4.209E+06	8.592E-04	3.643E+10
3	8.0000 h	0.000E+00	0.000E+00	2.423E+05	3.527E+09	0.000E+00	0.000E+00	2.058E+06	4.202E-04	4.329E+10
4	24.0000 h	0.000E+00	0.000E+00	2.222E+05	1.337E+10	0.000E+00	0.000E+00	1.177E+05	2.403E-05	3.907E+10
5	96.0000 h	0.000E+00	0.000E+00	1.502E+05	4.766E+10	0.000E+00	0.000E+00	3.009E-01	6.142E-11	2.370E+09
6	720.0000 h	0.000E+00	0.000E+00	4.865E+03	9.536E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	6.056E+03
Xe-133	TOTALS		0.000E+00		1.319E+14	2.528E+11				1.441E+11
Xe-135m	INITIAL	0.000E+00		1.480E+06				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	1.919E-02	5.563E+08	1.067E+06	1.483E+02	5.242E+00	1.070E-09	3.128E+04
2	4.0000 h	0.000E+00	0.000E+00	1.031E-02	9.097E+01	0.000E+00	0.000E+00	2.602E-02	5.311E-12	6.601E+03
3	8.0000 h	0.000E+00	0.000E+00	6.758E-03	1.209E+02	0.000E+00	0.000E+00	3.150E-03	6.430E-13	1.038E+02
4	24.0000 h	0.000E+00	0.000E+00	1.262E-03	1.887E+02	0.000E+00	0.000E+00	3.669E-05	7.490E-15	4.027E+01
5	96.0000 h	0.000E+00	0.000E+00	6.639E-07	4.331E+01	0.000E+00	0.000E+00	7.297E-14	1.490E-23	4.747E-01
6	720.0000 h	0.000E+00	0.000E+00	2.535E-35	2.279E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.440E-10
Xe-135m	TOTALS		0.000E+00		5.563E+08	1.067E+06				3.802E+04
Xe-135	INITIAL	0.000E+00		4.890E+08				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	4.200E+02	2.521E+11	4.837E+08	6.718E+04	1.078E+04	2.201E-06	4.204E+07
2	4.0000 h	0.000E+00	0.000E+00	3.607E+02	2.805E+06	0.000E+00	0.000E+00	6.547E+03	1.337E-06	6.113E+07
3	8.0000 h	0.000E+00	0.000E+00	2.661E+02	4.479E+06	0.000E+00	0.000E+00	2.414E+03	4.927E-07	5.965E+07
4	24.0000 h	0.000E+00	0.000E+00	7.878E+01	8.864E+06	0.000E+00	0.000E+00	4.456E+01	9.097E-09	3.418E+07
5	96.0000 h	0.000E+00	0.000E+00	3.291E-01	3.712E+06	0.000E+00	0.000E+00	7.038E-07	1.437E-16	6.430E+05
6	720.0000 h	0.000E+00	0.000E+00	7.908E-22	1.557E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.015E-02
Xe-135	TOTALS		0.000E+00		2.521E+11	4.837E+08				1.976E+08
I-131	INITIAL	0.000E+00		1.560E+09				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	1.549E+03	8.126E+11	7.796E+07	1.083E+04	1.859E+03	3.795E-07	7.087E+06
2	4.0000 h	0.000E+00	0.000E+00	1.538E+03	1.111E+07	0.000E+00	0.000E+00	1.305E+03	2.664E-07	1.127E+07
3	8.0000 h	0.000E+00	0.000E+00	1.516E+03	2.199E+07	0.000E+00	0.000E+00	6.428E+02	1.312E-07	1.347E+07
4	24.0000 h	0.000E+00	0.000E+00	1.431E+03	8.485E+07	0.000E+00	0.000E+00	3.786E+01	7.728E-09	1.230E+07
5	96.0000 h	0.000E+00	0.000E+00	1.105E+03	3.269E+08	0.000E+00	0.000E+00	1.105E-04	2.256E-14	7.700E+05
6	720.0000 h	0.000E+00	0.000E+00	1.175E+02	9.897E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.248E+00
I-131	TOTALS		0.000E+00		8.140E+11	7.796E+07				4.490E+07

TRAILS_PC -- Transport of Radioactive Material in Linear Systems, v1.0a
FHA - CR, EAB, & LPZ 100 h decay, 617 rods, SLCRS filt, no CR isol, CST [2x617.i ***PROGENY INGROWTH ON ***

STEP	TIME	N/A CURRENT uCi	N/A INTEGRD uCi-sec	CNMT CURRENT uCi	CNMT INTEGRD uCi-sec	RELEASED uCi	AVERAGE RELEASE uCi/sec	CONTROL ROOM CURRENT uCi	CONTROL ROOM CURRENT uCi/cc	CONTROL ROOM INTEGRD uCi-sec
I-132	INITIAL	0.000E+00		1.080E+09				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	5.911E+02	5.393E+11	5.174E+07	7.186E+03	9.460E+02	1.931E-07	3.937E+06
2	4.0000 h	0.000E+00	0.000E+00	3.235E+02	3.196E+06	0.000E+00	0.000E+00	3.660E+02	7.472E-08	4.398E+06
3	8.0000 h	0.000E+00	0.000E+00	9.691E+01	2.707E+06	0.000E+00	0.000E+00	5.479E+01	1.119E-08	2.360E+06
4	24.0000 h	0.000E+00	0.000E+00	7.803E-01	1.148E+06	0.000E+00	0.000E+00	2.752E-02	5.618E-12	4.153E+05
5	96.0000 h	0.000E+00	0.000E+00	2.943E-10	9.321E+03	0.000E+00	0.000E+00	3.924E-17	8.010E-27	2.087E+02
6	720.0000 h	0.000E+00	0.000E+00	0.000E+00	3.515E-06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.975E-13
I-132	TOTALS		0.000E+00	-	5.393E+11	5.174E+07				1.111E+07
I-133	INITIAL	0.000E+00		1.340E+08				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	1.254E+02	6.950E+10	6.668E+06	9.261E+02	1.546E+02	3.157E-08	5.950E+05
2	4.0000 h	0.000E+00	0.000E+00	1.173E+02	8.732E+05	0.000E+00	0.000E+00	1.023E+02	2.088E-08	9.120E+05
3	8.0000 h	0.000E+00	0.000E+00	1.026E+02	1.581E+06	0.000E+00	0.000E+00	4.473E+01	9.131E-09	1.002E+06
4	24.0000 h	0.000E+00	0.000E+00	6.022E+01	4.582E+06	0.000E+00	0.000E+00	1.637E+00	3.342E-10	7.505E+05
5	96.0000 h	0.000E+00	0.000E+00	5.467E+00	5.915E+06	0.000E+00	0.000E+00	5.619E-07	1.147E-16	2.851E+04
6	720.0000 h	0.000E+00	0.000E+00	5.091E-09	5.906E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.785E-03
I-133	TOTALS		0.000E+00		6.951E+10	6.668E+06				3.288E+06
I-135	INITIAL	0.000E+00		9.110E+04				0.000E+00		
1	2.0000 h	0.000E+00	0.000E+00	7.386E-02	4.677E+07	4.487E+03	6.232E-01	9.744E-02	1.989E-11	3.831E+02
2	4.0000 h	0.000E+00	0.000E+00	5.989E-02	4.798E+02	0.000E+00	0.000E+00	5.585E-02	1.140E-11	5.380E+02
3	8.0000 h	0.000E+00	0.000E+00	3.937E-02	7.044E+02	0.000E+00	0.000E+00	1.835E-02	3.746E-12	4.851E+02
4	24.0000 h	0.000E+00	0.000E+00	7.354E-03	1.099E+03	0.000E+00	0.000E+00	2.138E-04	4.364E-14	2.346E+02
5	96.0000 h	0.000E+00	0.000E+00	3.868E-06	2.523E+02	0.000E+00	0.000E+00	4.251E-13	8.678E-23	2.766E+00
6	720.0000 h	0.000E+00	0.000E+00	1.477E-34	1.328E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	5.500E-09
I-135	TOTALS		0.000E+00		4.677E+07	4.487E+03				1.644E+03

TRAILS_PC -- Transport of Radioactive Material in Linear Systems, v1.0a
FHA - CR, EAB, & LPZ 100 h decay, 617 rods, SLCRS filt, no CR isol, CST [2x617.i ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	SKIN-DE DOSE mrem	THY-CDE DOSE mrem	EDE DOSE mrem	SKIN-DE DOSE mrem	THY-CDE DOSE mrem	EDE DOSE mrem	EDE RATE DOSE RATE mrem/h	SKIN-DE DOSE mrem	THY-CDE DOSE mrem
Kr-85m										
2.0000 h	2.83E-04	5.59E-04	0.00E+00	1.37E-05	2.70E-05	0.00E+00	1.92E-07	1.74E-07	7.56E-06	0.00E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.55E-07	9.01E-08	1.00E-05	0.00E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.01E-07	2.42E-08	7.88E-06	0.00E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.35E-08	1.27E-10	2.89E-06	0.00E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.33E-10	6.98E-21	9.14E-09	0.00E+00
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.51E-21	0.00E+00	3.34E-19	0.00E+00
TOTALS	2.83E-04	5.59E-04	0.00E+00	1.37E-05	2.70E-05	0.00E+00	7.21E-07		2.83E-05	0.00E+00
Kr-85										
2.0000 h	3.40E+00	4.75E+02	0.00E+00	1.64E-01	2.29E+01	0.00E+00	2.54E-03	2.40E-03	7.07E+00	0.00E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.06E-03	1.70E-03	1.13E+01	0.00E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.90E-03	8.48E-04	1.36E+01	0.00E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.58E-03	5.29E-05	1.28E+01	0.00E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.83E-04	2.00E-10	5.09E-01	0.00E+00
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.61E-10	0.00E+00	1.28E-06	0.00E+00
TOTALS	3.40E+00	4.75E+02	0.00E+00	1.64E-01	2.29E+01	0.00E+00	1.63E-02		4.52E+01	0.00E+00
Xe-131m										
2.0000 h	4.19E+00	4.65E+01	0.00E+00	2.03E-01	2.25E+00	0.00E+00	3.13E-03	2.96E-03	6.90E-01	0.00E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.99E-03	2.08E-03	1.10E+00	0.00E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.97E-03	1.03E-03	1.32E+00	0.00E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.50E-03	6.18E-05	1.21E+00	0.00E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.11E-04	1.96E-10	4.65E-02	0.00E+00
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.46E-10	0.00E+00	9.83E-08	0.00E+00
TOTALS	4.19E+00	4.65E+01	0.00E+00	2.03E-01	2.25E+00	0.00E+00	1.98E-02		4.37E+00	0.00E+00
Xe-133m										
2.0000 h	2.78E+01	1.85E+02	0.00E+00	1.34E+00	8.93E+00	0.00E+00	2.06E-02	1.94E-02	2.73E+00	0.00E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.24E-02	1.34E-02	4.29E+00	0.00E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.77E-02	6.34E-03	4.98E+00	0.00E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.22E-02	3.20E-04	4.27E+00	0.00E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.03E-03	4.68E-10	1.36E-01	0.00E+00
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E-09	0.00E+00	1.33E-07	0.00E+00
TOTALS	2.78E+01	1.85E+02	0.00E+00	1.34E+00	8.93E+00	0.00E+00	1.24E-01		1.64E+01	0.00E+00

TRAILS_PC -- Transport of Radioactive Material in Linear Systems, v1.0a
FHA - CR, EAB, & LPZ 100 h decay, 617 rods, SLCRS filt, no CR isol, CST [2x617.i ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	SKIN-DE DOSE mrem	THY-CDE DOSE mrem	EDE DOSE mrem	SKIN-DE DOSE mrem	THY-CDE DOSE mrem	EDE DOSE mrem	EDE RATE DOSE RATE mrem/h	SKIN-DE DOSE mrem	THY-CDE DOSE mrem
Xe-133										
2.0000 h	1.76E+03	3.34E+03	0.00E+00	8.53E+01	1.61E+02	0.00E+00	1.31E+00	1.24E+00	4.95E+01	0.00E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.09E+00	8.68E-01	7.85E+01	0.00E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.48E+00	4.24E-01	9.33E+01	0.00E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.24E+00	2.43E-02	8.42E+01	0.00E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.14E-02	6.20E-08	3.06E+00	0.00E+00
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.39E-07	0.00E+00	5.22E-06	0.00E+00
TOTALS	1.76E+03	3.34E+03	0.00E+00	8.53E+01	1.61E+02	0.00E+00	8.20E+00		3.08E+02	0.00E+00
Xe-135m										
2.0000 h	9.08E-02	2.89E-02	0.00E+00	4.39E-03	1.40E-03	0.00E+00	2.18E-05	1.32E-05	1.38E-04	0.00E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.61E-06	6.54E-08	2.92E-05	0.00E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.25E-08	7.92E-09	4.59E-07	0.00E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.81E-08	9.22E-11	1.78E-07	0.00E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.99E-10	1.83E-19	1.26E-09	0.00E+00
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.64E-19	0.00E+00	1.67E-18	0.00E+00
TOTALS	9.08E-02	2.89E-02	0.00E+00	4.39E-03	1.40E-03	0.00E+00	2.65E-05		1.68E-04	0.00E+00
Xe-135										
2.0000 h	2.40E+01	3.91E+01	0.00E+00	1.16E+00	1.89E+00	0.00E+00	1.71E-02	1.58E-02	5.55E-01	0.00E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.49E-02	9.60E-03	8.08E-01	0.00E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.43E-02	3.54E-03	7.88E-01	0.00E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.39E-02	6.54E-05	4.52E-01	0.00E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.57E-04	1.03E-12	5.10E-03	0.00E+00
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.66E-12	0.00E+00	5.37E-11	0.00E+00
TOTALS	2.40E+01	3.91E+01	0.00E+00	1.16E+00	1.89E+00	0.00E+00	8.04E-02		2.61E+00	0.00E+00
I-131										
2.0000 h	5.90E+00	3.11E+00	3.65E+04	2.85E-01	1.50E-01	1.76E+03	4.40E-03	4.16E-03	4.62E-02	5.42E+02
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.00E-03	2.92E-03	7.35E-02	8.62E+02
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.36E-03	1.44E-03	8.78E-02	1.03E+03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.64E-03	8.47E-05	8.02E-02	9.41E+02
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.87E-04	2.47E-10	3.01E-03	3.53E+01
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.58E-10	0.00E+00	5.86E-09	6.88E-05
TOTALS	5.90E+00	3.11E+00	3.65E+04	2.85E-01	1.50E-01	1.76E+03	2.77E-02		2.91E-01	3.41E+03

TRAILS_PC -- Transport of Radioactive Material in Linear Systems, v1.0a
FHA - CR, EAB, & LPZ 100 h decay, 617 rods, SLCRS filt, no CR isol, CST [2x617.i ***PROGENY INGROWTH ON ***

	- EXCLUSION AREA BOUNDARY -			--- LOW POPULATION ZONE ---			------------ CONTROL ROOM ------------			
	EDE DOSE mrem	SKIN-DE DOSE mrem	THY-CDE DOSE mrem	EDE DOSE mrem	SKIN-DE DOSE mrem	THY-CDE DOSE mrem	EDE DOSE mrem	EDE RATE DOSE RATE mrem/h	SKIN-DE DOSE mrem	THY-CDE DOSE mrem
I-132										
2.0000 h	2.44E+01	7.22E+00	1.45E+02	1.18E+00	3.49E-01	6.98E+00	1.53E-02	1.32E-02	8.97E-02	1.80E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.70E-02	5.11E-03	1.00E-01	2.01E+00
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.15E-03	7.64E-04	5.38E-02	1.08E+00
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.61E-03	3.84E-07	9.47E-03	1.89E-01
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.85E-07	5.47E-22	2.85E-06	5.71E-05
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.61E-22	0.00E+00	2.71E-21	5.43E-20
TOTALS	2.44E+01	7.22E+00	1.45E+02	1.18E+00	3.49E-01	6.98E+00	4.31E-02		2.53E-01	5.07E+00
I-133										
2.0000 h	8.10E-01	7.53E-01	5.21E+02	3.92E-02	3.64E-02	2.52E+01	5.93E-04	5.55E-04	1.10E-02	7.59E+00
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.10E-04	3.67E-04	1.68E-02	1.16E+01
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.99E-04	1.61E-04	1.85E-02	1.28E+01
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.48E-04	5.88E-06	1.38E-02	9.57E+00
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.71E-05	2.02E-12	3.15E-04	2.18E-01
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.90E-12	0.00E+00	7.21E-11	4.99E-08
TOTALS	8.10E-01	7.53E-01	5.21E+02	3.92E-02	3.64E-02	2.52E+01	3.27E-03		6.04E-02	4.18E+01
I-135										
2.0000 h	1.48E-03	4.49E-04	6.09E-02	7.17E-05	2.17E-05	2.94E-03	1.04E-06	9.52E-07	6.26E-06	8.49E-04
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.46E-06	5.45E-07	8.79E-06	1.19E-03
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.32E-06	1.79E-07	7.92E-06	1.08E-03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.36E-07	2.09E-09	3.83E-06	5.20E-04
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.50E-09	4.15E-18	2.71E-08	3.68E-06
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.97E-18	0.00E+00	3.59E-17	4.88E-15
TOTALS	1.48E-03	4.49E-04	6.09E-02	7.17E-05	2.17E-05	2.94E-03	4.46E-06		2.68E-05	3.64E-03
ALL NUCLIDES										
2.0000 h	1.86E+03	4.09E+03	3.72E+04	8.96E+01	1.98E+02	1.80E+03	1.38E+00	1.30E+00	6.06E+01	5.52E+02
4.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.18E+00	9.03E-01	9.62E+01	8.76E+02
8.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.57E+00	4.39E-01	1.14E+02	1.04E+03
24.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.30E+00	2.49E-02	1.03E+02	9.51E+02
96.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.33E-02	6.32E-08	3.76E+00	3.56E+01
720.0000 h	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.41E-07	0.00E+00	6.74E-06	6.88E-05
TOTALS	1.86E+03	4.09E+03	3.72E+04	8.96E+01	1.98E+02	1.80E+03	8.51E+00		3.78E+02	3.46E+03

ATTACHMENT D

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 155

Applicable Draft UFSAR Changes

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 dose calculation methodology. Section C.3

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.

for which releases are assumed to occur over a two hour period

For Condition IV DBAs which do not initiate a CIB signal, the accident duration is 8 hours (except for the fuel handling accident (FHA) and the locked rotor accident (LRA) for which releases are assumed to continue over a 30-day period).

The information and data required to develop the radiological consequences for the main control room are presented in the respective sections describing the design basis accident analysis.

The main control room dose presented in Table 15.0-13 has been calculated to be less than the limit specified in General Design Criterion 19 and the main control room may, therefore, be safely occupied during any condition of operation.

6.4.4.2 Toxic Gas Protection

The main control room design provides protection of the personnel in the main control room from any toxic effects from spills of chemicals stored onsite. The effects of spills of chemicals along transportation routes are evaluated in Section 2.2.3.

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 compressed air for 1 hour, after which, the main control room will remain isolated for the duration of the accident.

Following the first hour, self-contained breathing apparatus units and sufficient reserve air cylinders are available to support the minimum control room shift composition for at least five additional hours. This satisfies Regulatory Guide 1.78 and 1.95. Sufficient additional units are provided to support the members of the emergency squad stationed outside the control room for one hour, after which these personnel would move away from the area affected by the toxic release. Air cylinders brought from off-site locations may be used to extend capacity beyond six hours.

Redundant, sensitive, and automatic Seismic Category I detection and isolation equipment is provided for the detection of chlorine gas.

The storage areas of toxic gases and chemicals that could produce toxic gases are shown in Table 6.4-3 and on Figure 6.4-5.

6.4.5 Inspection and Testing Requirements

The major items of equipment that maintain the habitability of the main control room are the emergency supply filtration units, their fans, 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

containment with open penetrations and

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 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 11.1-1

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

Isotope	Core (Ci)	Fraction of Core Activity in Gap	Fuel Rod Gap Activity (Ci)
I-131	6.9×10^{7}	0.1	6.9×10^{6}
I-132	9.9×10^{7}	0.1	9.9×10^{6}
I-133	1.6×10^{8}	0.1	1.6×10^{7}
I-134	1.8×10^{8}	0.1	1.8×10^{7}
I-135	1.4×10^{8}	0.1	1.4×10^{7}
Kr-83m	1.2×10^{7}	0.1	1.2×10^{6}
Kr-85m	3.0×10^{7}	0.1	3.0×10^{6}
Kr-85	6.8×10^{5}	0.1~~**~~	6.8×10^{4}
Kr-87	5.9×10^{7}	0.1	5.9×10^{6}
Kr-88	8.3×10^{7}	0.1	8.3×10^{6}
Kr-89	1.1×10^{8}	0.1	1.1×10^{7}
Xe-131m	4.2×10^{5}	0.1	4.2×10^{4}
Xe-133m	3.7×10^{6}	0.1	3.7×10^{5}
Xe-133	1.6×10^{8}	0.1	1.6×10^{7}
Xe-135m	4.2×10^{7}	0.1	4.2×10^{6}
Xe-135	4.1×10^{7}	0.1	4.1×10^{6}
Xe-137	1.4×10^{8}	0.1	1.4×10^{7}
Xe-138	1.4×10^{8}	0.1	1.4×10^{7}

NOTES:

*Based on 650 days of operation at 2,766 MWt, for use in Chapter 15.

~~**In fuel handling accident analysis, the fraction used for Kr-85 is 0.3, in accordance with Regulatory Guide 1.25.~~

Refer to Table 15.7-6 for activities used in the fuel handling accident.

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		10.5	1.2×10^{-2}	4.6×10^{-3}	1.5	1.4×10^{-3}	6.1×10^{-4}
Concurrent Iodine spike		9.1	2.2×10^{-2}	6.7×10^{-3}	3.2	6.8×10^{-3}	2.2×10^{-3}
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	3.25×10^{1}	3.41	2.09	1.44×10^{1}	3.48×10^{-1}	2.17×10^{-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 ~~2.9x10^1~~	1.9E+00 ~~2.33~~	~~6.58~~	1.8E+00 ~~1.4~~	<1E-1 ~~1.1x10^-1~~	~~3.2x10^-1~~
		Thyroid CDE	EDE	Skin DE	Thyroid CDE	EDE	Skin DE
Small Break LOCA	15.6.5.5	2.9×10^{2}	2.2	1.3	2.0×10^{1}	1.3×10^{-1}	7.7×10^{-2}

NOTE:

*For duration of accident

TABLE 15.0-13

Control Room Doses, rem, From Design Basis Accidents[5,7]

Accident	Thyroid	Gamma	Beta	Notes
Main Steam Line Break				
Co-incident Spike	11.0	6.8E-4	4.8E-3	3
Pre-incident Spike	5.4	1.8E-4	1.7E-3	3
Small Line Break	8.1	8.0E-4	7.7E-3	3
Steam Generator Tube Rupture				
Co-incident Spike	1.9	3.0E-4	6.1E-3	3
Pre-incident Spike	8.7	5.0E-4	7.9E-3	3
Rod Ejection Accident	4.9	4.9E-4	3.8E-3	3
Fuel Handling Accident	3.5E+00 ~~2.3~~	<2E-01 ~~9.3E-3~~	<1E+00 ~~5.3E-1~~	3
Locked Rotor Accident	1.1	1.1E-2	1.5E-1	1,4
Loss of Auxiliary AC Power	2.1	1.8E-4	1.2E-2	3
Waste Gas System Rupture				
Line Break	---	5.8E-2	1.3	3
Tank Rupture	---	3.5E-2	9.7	3
DBA LOCA	1.3	3.2E-1	1.2E-1	2

	Thyroid CDE	EDE	Skin DE
Small Break LOCA[6]	11.0	3.2E-3	3.2E-2

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 a minimum of 30 minutes at 20,000 cfm at no later than T=8 hr post-accident initiation.
5: Reference: ERS-SFL-93-004
6: Reference: ERS-SFL-94-014
7: Listed dose values represent the bounding value which may be higher than current analysis results.

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. ~~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 11.5).~~ <Insert>

The fuel handling accident sequence other of events consists of the dropping of one fuel assembly on ~~another~~ fuel assembl~~y~~ies ~~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 100 hours and peak inventories for the damaged fuel assemblies. The fuel pool water provides retention capabilities for radioiodines as described in Table 15.7-6.

The radioactivity released from the ~~fuel~~ pool into the ~~fuel~~ building atmosphere is filtered by the supplementary leak collection and release system (Section 6.5.1).

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 1.25, 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).

~~To verify that a fuel handling accident 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, has been completed. The time required for air to travel from the radiation monitor to the first containment isolation valve is greater than the

<Page 15.7-4a Section 15.7.4.2.1 Insert>

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.

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.

<Insert>

<Page 15.7-4 Section 15.7.4.1 Insert>

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.

15.7.4.3 Radiological Consequences

⟨Insert⟩

~~A fuel handling accident is defined as the dropping of one spent fuel assembly onto another fuel assembly in the spent fuel storage area. The accident is postulated to cause damage to all of the fuel rods in the dropped assembly plus an additional 50 rods in the struck fuel assembly with subsequent release of all the activity in the fuel rod gap.~~ The gap activity in the core fuel assemblies consists of 10 percent of the core noble gas and iodine activities, except for Kr-85, which is taken as 30 percent of the Kr-85 core activity and I-131, which is taken as 12% 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 ~~fuel~~ pool where 100 percent of the noble gas and 1.0 percent of the iodine, is then released into the building. ~~Even though the~~ The analysis assumes that activity, which is released ~~leaks~~ into the ~~fuel~~ building atmosphere is ~~may be~~ exhausted through filters over a 2-hour period, ~~the analysis is performed assuming the release is instantaneous~~. 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.

The radiological consequences of the postulated fuel handling accident are analyzed based on the assumptions and initial maximum assembly gap activities listed in Table 15.7-6. ~~with the initial core gap activities given in Table 15.0-7. The resulting releases are shown in Table 15.7-7.~~ Offsite doses are calculated using the ~~preceding~~ releases in combination with the atmospheric dispersion values given in Table 15.0-~~11~~ 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, are well within the guidelines of 10 CFR 100, that is, less than 75 Rem thyroid and 6 Rem whole body.

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.

<Page 15.7-5, Section 15.7.4.3 Insert>

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 in the fuel building is 50 rods, and in the reactor containment building, 353 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.

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
~~DLC~~ ~~1989.~~ Unit 2 Fuel Handling Accident Doses at EAB, LPZ, Common Control Room. ERS-SFL-89-019, revision 2.

TABLE 15.7-6

ASSUMPTIONS USED FOR THE
FUEL HANDLING ACCIDENT ANALYSIS

Power level (MWt)	~~2,766~~ 2,705
Operating time (days)	~~650~~ 1,500
~~Gap activity~~	~~Table 15.0-7~~
Minimum time since shutdown (hrs)	100
Total number of fuel assemblies in core	157
Number of fuel rods per assembly	264
Fuel damage* (rods)	~~1 assembly and 50 fuel rods~~ 617
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
Noble gases	1.0
Iodine fraction above pool	
Inorganic	0.75
Organic	0.25
Fuel building filter efficiency (%)	
Inorganic	95
Organic	95
~~Type of release~~	~~puff~~
Release duration (hrs)	2

<Insert footnote>

<Table 15.7-6 Insert>

*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.

<Addition - Insert>

TABLE 15.7-6 (Cont)

ASSUMPTIONS 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
Kr-85	6.60E+03	3.27E+03
Xe-131m	6.41E+03	1.06E+03
Xe-133m	1.22E+04	2.02E+03
Xe-133	6.56E+05	1.08E+05
Xe-135m	3.85E+00	6.35E-01
Xe-135	1.27E+03	2.09E+02
I-131	3.38E+05	6.69E+04
I-132	2.81E+05	4.64E+04
I-133	3.47E+04	5.73E+03
I-135	2.36E+01	3.90E+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).

TABLE 15.7-7

FUEL HANDLING ACCIDENT IN THE FUEL BUILDING
RELEASES TO THE ENVIRONMENT

Nuclides	Releases (Ci) 0-2 hr
Kr-83m	2.0×10^{-8}
Kr-85m	5.4×10^{-3}
Kr-85	2.6×10^{3}
Kr-88	1.6×10^{-6}
Xe-131m	5.0×10^{2}
Xe-133m	2.1×10^{3}
Xe-133	1.4×10^{5}
Xe-135m	1.8
Xe-135	2.6×10^{2}
I-131	3.8×10^{1}
I-132	2.6×10^{1}
I-133	3.4
I-135	2.8×10^{-3}

Delete - included by reference.

QADMOD

Program QADMOD calculates dose rates at a series of detector locations with shielding for a number of different source points representing volumetric sources. The program is a modified version of the QAD P-5 program written at the Los Alamos Scientific Laboratory by R. E. Malenfant. This program has been upgraded to include: 1) the FASTER geometry routines, 2) a point source option, 3) a translated cylindrical source volume option, and 4) internal library data for conversion factors, build-up factor coefficients, and mass attenuation factors for several materials and compositions.

Insert

15A.1 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.

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TRAILS_PC

Program TRAILS_PC performs calculations involving transport of radioactive species between compartments that are related by first order linear processes. It is specifically structured to evaluate the transport of radioactivity in design basis accidents, and for calculating dose rates and doses at a user defined offsite location and in the control room. This code was developed and tested at BVPS and has been benchmarked against the SWEC DRAGON code.

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.

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, DC, External Dose-Rate Conversion Factors for Calculation of Dose to the Public, DOE/EH-0070, 1988

Eckerman, K. F., etal, 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

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

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

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

DLC 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