FENOC
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077-0004

Lew W. Myers
Senior Vice President

724-682-5234
Fax: 724-643-8069

October 29, 2001
L-01-113

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

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
License Amendment Request Nos. 281 and 152

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above licenses in the form of changes to the technical specifications (TSs). The proposed amendment revises the TS 3.9.3, "Refueling Operations – Decay Time," decay time of 150 hours to 100 hours.

The changes proposed herein were originally submitted to the NRC for review and approval as part of FENOC letter L-01-038, dated March 19, 2001. Letter L-01-038 transmitted License Amendment Request (LAR) Nos. 219 and 73 that proposed changes to the Beaver Valley Power Station (BVPS) fuel handling accident (FHA) analyses and containment closure requirements. The new FHA analyses use a decay time of 100 hours as the basis for radiological dose assessment. The new analyses form the basis for the reduction of the TS 3.9.3 decay time from 150 hours to 100 hours.

During the NRC staff review of LAR Nos. 219 and 73, issues were raised regarding how the proposed reduction of the TS 3.9.3 decay time from 150 hours to 100 hours may impact spent fuel pool performance at BVPS. Specifically, the NRC staff requested additional information related to the impact of this reduction in decay time on the hydrothermal performance of BVPS spent fuel pool systems. In response to the NRC issues and in order to facilitate the timely approval of LAR Nos. 219 and 73, FENOC letter L-01-094, dated July 6, 2001, submitted a partial withdrawal of LAR Nos. 219 and 73. This partial withdrawal requested that the proposed change to TS 3.9.3, "Refueling Operations – Decay Time," decay time from 150 hours to 100 hours be withdrawn pending further evaluation and resolution of spent fuel pool performance issues at BVPS.

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The NRC review and approval of LAR Nos. 219 and 73 has been completed and the corresponding Amendment Nos. 241 and 121 were issued on August 30, 2001. These amendments incorporate the FHA radiological analyses based on a 100-hour decay time into the current licensing basis for BVPS Unit No. 1 and 2. Additionally, FENOC has completed supporting spent fuel pool performance analyses that are provided with this submittal for NRC consideration in the review and approval of this request.

Proposed TS changes for Unit No. 1 are presented in Attachment A-1. Proposed TS changes for Unit No. 2 are presented in Attachment A-2. The safety analysis (including the no significant hazards evaluation) is presented in Attachment B. Responses to the NRC request for additional information is provided in Attachment C. Graphical representations of the analyses of spent fuel pool performance, based on decay time and component cooling water temperature, is presented in Attachment D-1 and D-2 for BVPS Unit Nos. 1 and 2, respectively. Attachment E provides a new commitment made in this submittal. Proposed changes to the TS Bases are provided as Attachments F-1 and F-2 for informational purposes only for Unit Nos. 1 and 2, respectively.

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

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

The BVPS Unit No. 2 portion of this change (LAR No. 152) is requested to be approved by February 1, 2002, in order to support the ninth refueling outage (2R09).

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 29, 2001.

Sincerely,

Lew W. Myers

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Proposed TS changes for Unit No. 1 are presented in Attachment A-1. Proposed TS changes for Unit No. 2 are presented in Attachment A-2. The safety analysis (including the no significant hazards evaluation) is presented in Attachment B. Responses to the NRC request for additional information is provided in Attachment C. Graphical representations of the analyses of spent fuel pool performance, based on decay time and component cooling water temperature, is presented in Attachment D-1 and D-2 for BVPS Unit Nos. 1 and 2, respectively. Attachment E provides a new commitment made in this submittal. Proposed changes to the TS Bases are provided as Attachments F-1 and F-2 for informational purposes only for Unit Nos. 1 and 2, respectively.

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Lew W. Myers

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

bc: A. J. Dometrovich
R. W. Fedin
M. E. O'Reilly
J. J. Raleigh
S. J. Sarver
B. F. Sepelak
M. F. Testa
Central File - *Keywords: LAR, Fuel Handling Accident, Spent Fuel Pool, Decay Time*

ATTACHMENT A-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 281

The following is a list of the affected pages:

3/4 9-3

REFUELING OPERATIONS

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

ATTACHMENT A-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 152

The following is a list of the affected pages:

3/4 9-3

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION OF OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 281 and 152

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 281 and 152
REVISION OF TECHNICAL SPECIFICATION 3/4.9.3 DECAY TIME

A. DESCRIPTION OF AMENDMENT REQUEST

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the Beaver Valley Power Station (BVPS), Unit No. 1 and No. 2, facility operating licenses DPR-66 and NPF-73, respectively, in the form of changes to the technical specifications (TSs). The limiting condition for operation (LCO) for TS 3.9.3, "Refueling Operations - Decay Time," and the associated surveillance requirement (SR) 4.9.3 will be revised by replacing "150 hours" with "100 hours". The amendment requests also include administrative, editorial, and format changes.

B. DESIGN BASES

The basis for the TS 3/4.9.3 specified minimum requirement for reactor subcriticality prior to the 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 that could be released in a fuel handling accident (FHA). A FHA is classified as an American Nuclear Society Condition IV event, faults that are not expected to occur but are postulated because their consequences include the potential for the release of significant amounts of radioactive material. The FHA is postulated to occur in the fuel building and in containment.

A FHA is defined as the dropping of one spent fuel assembly onto another fuel assembly in the spent fuel storage area or in the containment building. Currently, the most limiting FHA is postulated to cause fuel damage with subsequent release of all the activity in the fuel rod gap. The gap inventory released into the pool containing irradiated fuel for the total of 137 fuel rods is based on 100 hours of decay resulting from the time between shutdown and movement of the first fuel assembly. The total number of 137 damaged fuel rods and the 100 hours of decay time are applicable to both units.

The bases for the failure of the postulated 137 rods for a FHA in the fuel building and the radiological analyses based on 100 hours of decay time are approved in the NRC safety evaluation reports (SERs) for Amendment Nos. 241 and 121, issued on August 30, 2001. The SERs in support of these amendments are consistent with the guidance contained in NUREG-0800, "Standard Review Plan [SRP]," Section 15.7.4, "Radiological Consequences of Fuel Handling Accident;" Regulatory Guide (RG) 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," with the exceptions of iodine filter efficiencies which follow the guidance in RG 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants;" 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, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," (USNRC 1988; Section 3.2.2).

The administrative changes are not related to any technical design basis.

C. JUSTIFICATION

The Beaver Valley Power Station (BVPS) current licensing basis (CLB) FHA radiological analyses are based on a decay time, resulting from the time between shutdown and movement of the first fuel assembly, of 100 hours for BVPS Unit Nos. 1 and 2. The bases for the decay time specification is to ensure that sufficient time has elapsed to allow the radioactive decay of short lived fission products. The proposed change in the TS 3/4.9.3 decay time from 150 hours to 100 hours is consistent with the bases for the decay time specification and the assumptions used in the accident analysis. The radiological analyses approved as part of Amendment Nos. 241 and 121 demonstrate that should a FHA occur within the containment or the fuel building that involves irradiated fuel with at least 100 hours of decay, the projected offsite doses for this event will be well within the applicable regulatory limits of 10 CFR Part 50.67 of 25 rem total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) (for any 2-hour period), and low population zone (LPZ) (for the entire period of radioactive cloud passage), and are less than the more restrictive guidance criteria in the SRP Section 15.0.1 and RG 1.183, both entitled, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," of 6.3 rem TEDE for the 2 hour release duration. Control room personnel doses (for the duration of the accident) are less than the 10 CFR Part 50.67 limit of 5 rem TEDE.

The change in decay time will result in an increase in the spent fuel pool heat load. BVPS has evaluated the effects of an increased heat load on the spent fuel pool cooling system due to conducting a core offload at 100 hours. The ability to conduct a core offload at 100 hours, or between 100 and 150 hours, is dependent on the results of this evaluation, which are graphically represented in Attachments D-1 and D-2 for Unit Nos. 1 and 2, respectively. FENOC commits to establish administrative controls to control decay times prior to the movement of irradiated fuel assemblies from the reactor core based on these graphs.

Therefore, based on the above, the change in decay time specified in Specification 3/4.9.3 from 150 hours to 100 hours is acceptable.

The proposed administrative, editorial, and format changes do not affect plant safety.

D. SAFETY ANALYSIS

Based on the BVPS CLB radiological analysis for a FHA using a 100-hour decay time and without crediting the use of the filtered SLCRS or the isolation of the containment, the resultant radiological consequences of this event will be well within the applicable regulatory limits of 10 CFR Part 50.67 of 25 rem TEDE at the EAB (for any 2-hour period) and LPZ (for the entire period of radioactive cloud passage), and are less than the more restrictive guidance criteria in the SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," and RG 1.183 of 6.3 rem TEDE for the 2 hour release duration. Control room personnel doses (for the duration of the accident) are less than the 10 CFR Part 50.67 limit of 5 rem TEDE. This radiological analysis is based on all airborne activity reaching the applicable building (fuel building or containment building) atmosphere being released to the environment over a 2 hour period. The 2-hour release period is based on the guidance contained in Regulatory Guide 1.183.

LCO 3.9.10, "Water Level – Reactor Vessel," and LCO 3.9.11, "Storage Pool Water Level," will continue to ensure that at least 23 feet of water is maintained over stored/seated fuel assemblies during fuel movement. LCO 3.9.3, "Refueling Operations - Decay Time," will continue to ensure that irradiated fuel is not moved in the reactor pressure vessel until at least 100 hours after shutdown. These LCOs will continue to ensure that two of the key assumptions used in the radiological safety analysis for a FHA are met.

A FHA is the only event during Core Alterations that is postulated to result in fuel damage and radiological release. The accidents that are postulated to occur during

Core Alterations, in addition to a FHA, are: inadvertent criticality (due to a control rod removal error or continuous control rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage.

The radiological consequences of the Core Alteration events other than the FHA remain unchanged. These events do not result in fuel cladding integrity damage. A radioactive release to the environment is not postulated since the activity is contained in the fuel rods. Therefore, the affected containment systems and minimum water level over fuel assemblies are not required to mitigate a radioactive release to the environment due to these Core Alteration events.

The proposed changes to the technical specification requirements will continue to ensure that the necessary plant equipment is operable in the plant conditions where these systems are required to operate to mitigate a DBA. The various administrative changes will continue to ensure that plant systems are available to support the assumptions of plant safety analysis and do not affect plant safety.

Therefore, based on the above, the change in decay time specified in Specification 3/4.9.3 from 150 hours to 100 hours is considered safe.

E. NO SIGNIFICANT HAZARDS EVALUATION

The proposed amendment revises the technical specification (TS) 3/4.9.3, "Refueling Operations – Decay Time," decay time requirement and associated surveillance requirements from 150 hours to 100 hours. The proposed amendment also includes administrative, editorial, and format changes to the Specifications.

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?

   No. The proposed change does not alter the manner in which fuel assemblies are handled or core alterations are performed. The proposed change does not alter the manner in which heavy loads are controlled at BVPS. The proposed change does not result in changes being made to structures, systems, or components (SSCs), or to event initiators or precursors. Also, the proposed change does not impact the design of plant systems such that previously analyzed SSCs would now be more likely to fail. The initiating conditions and assumptions for accidents described in the Updated Final Safety Analysis Report (UFSAR) remain as previously analyzed. Thus, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   The proposed revision of the decay time from 150 hours to 100 hours is consistent with the assumptions used in the NRC approved fuel handling accident (FHA) analyses for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2. The BVPS radiological analyses demonstrates that should a FHA occur within the containment or the fuel building that involves irradiated fuel with at least 100 hours of decay, the projected offsite doses for this event will be well within the applicable regulatory limits.

   Limiting Condition for Operation (LCO) 3.9.3, "Refueling Operations - Decay Time," will continue to ensure that irradiated fuel is not moved in the reactor pressure vessel until at least 100 hours after shutdown which is consistent with the FHA radiological analysis. This LCO will continue to ensure that key assumptions used in the radiological safety analysis are met. The previously analyzed SSCs are unaffected by the proposed change and continue to provide assurance that they are capable of performing their intended design function in mitigating the effects of design basis accidents (DBAs). As such, the consequences of accidents previously evaluated in the UFSAR will not be increased and no additional radiological source terms are generated. Therefore, there will be no reduction in the capability of those SSCs in limiting the radiological consequences of previously evaluated accidents and reasonable assurance that there is no undue risk to the health and safety of the public will continue to be provided. Thus, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

The proposed administrative, editorial, and format changes do not affect the probability or consequences of any accident.

Therefore, the proposed amendment does not significantly increase the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed amendment does not affect a previously evaluated accident; e.g., FHA. The proposed amendment takes credit for the normal decay of irradiated fuel and the existing radiological analyses for FHAs.

   The proposed change does not involve physical changes to analyzed SSCs or changes to the modes of plant operation defined in the technical specification. The proposed change does not involve the addition or modification of plant equipment (no new or different type of equipment will be installed) nor does it alter the design or operation of any plant systems. No new accident scenarios, accident or transient initiators or precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change.

   The proposed change does not cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new or different mode of failure has been created and no new or different equipment performance requirements are imposed for accident mitigation. As such, the proposed change has no effect on previously evaluated accidents.

The proposed administrative, editorial, and format changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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?

No. The proposed revision of the decay time from 150 hours to 100 hours is consistent with the assumptions used in the NRC approved FHA accident analyses for BVPS Unit Nos. 1 and 2 and thus does not involve a significant reduction in a margin of safety.

The proposed amendment does not alter the manner in which fuel assemblies are handled or core alterations are performed. The proposed amendment does not alter the manner in which heavy loads are controlled at BVPS.

The proposed changes to the TS requirements will continue to ensure that the necessary plant equipment is operable in the plant conditions where these systems are required to operate to mitigate a DBA. The proposed administrative, editorial, and format changes do not affect plant safety.

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 consistent with examples where there is no impact on previously analyzed accidents in the current licensing and design basis of the facility.

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.

ATTACHMENT C

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 281 and 152

Responses to NRC Request for Additional Information

Attachment C
Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 281 and 152

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
CHANGE IN TECHNICAL SPECIFICATION 3/4.9.3 DECAY TIME
BEAVER VALLEY POWER STATION UNIT NOS. 1 AND 2
(LICENSE AMENDMENT REQUEST NOS. 219 AND 73, AND 281 AND 152)

On May 23, 2001, the NRC transmitted an informal request for additional information (RAI) related to the review of FirstEnergy Nuclear Operating Company (FENOC) license amendment requests (LARs) 219 and 73. The NRC RAI and the FENOC responses are provided below:

"RAI for Beaver Valley 1&2 - Change in decay time specified in TS 3/4.9.3.

In your submittal of March 19, 2001, you request a change to TS 3.9.3, "Decay Time" to decrease the amount of time fuel must remain in the reactor vessel after shutdown before offloading. The change would reduce the decay time from 150 hours to 100 hours. On page B-38 of your submittal, you state that the change in decay time will result in an increase in the spent fuel pool (SFP) heat load. You also state that BVPS [Beaver Valley Power Station] will evaluate the effects of an increased heat load on the SFP cooling system due to conducting a core offload at 100 hours.

The impact of the increased heat load on the SFP is information we need to be able to fully evaluate your request to change the decay time in TS [technical specification] 3.9.3. Please submit the results of all evaluations performed on the impact of the increased heat load on the SFP and supporting systems. Your evaluation of the spent fuel cooling system should address both the planned and unplanned offload conditions. The use of the terminology "planned" and "unplanned" has been used by the staff for the review of SFP heat load changes since questions arose in the mid-1990's regarding refueling practices at Millstone Unit 1. A planned offload is a scheduled offload for refueling, maintenance, or decommissioning purposes. An unplanned offload is a previously unscheduled offload in response to an event or equipment failure. This difference in terminology was made to ensure SFP temperature evaluations accurately reflected actual licensee practices.

Your analyses should reflect the following:"

NRC RAI Question 1

As you have performed full core offloads during all your refueling outages, your planned offload is a full core offload. Therefore, Analysis Cases 1a. and 1b. should assume the offloading of a full core with all other storage locations filled.

FENOC Response

The current licensing basis (CLB) for the spent fuel pool offload analyses for BVPS Unit No. 1, which was licensed prior to the requirements of NUREG-0800, "Standard Review Plan," includes four cases for study. These cases are defined as follows:

Case 1a: A normal refueling load (72 fuel assemblies) of spent fuel assemblies discharged after 150 hours of decay time, with one train of the SFP cooling system operating,

Case 1b: Defined as being the same as Case 1a with both trains of SFP cooling operating,

Case 2: A full core discharged to the SFP after 150 hours of decay time, with both SFP cooling trains operating, and

Case 3: A refueling load discharged (72 fuel assemblies) discharged to the SFP after 150 hours of decay time, then a full core offload 60 days later with both SFP cooling trains operating.

The CLB for the spent fuel pool offload analyses for BVPS Unit No. 2, which was licensed under the guidance contained in the Standard Review Plan, includes four slightly different cases for study. These cases are defined as follows:

Case 1a: A normal refueling load (1/3 of a core) of spent fuel assemblies discharged after 150 hours of decay time, plus 1/3 of a core with 400 days of decay, with one train of the SFP cooling system operating,

Case 1b: Defined as being the same as Case 1 with both trains of SFP cooling operating,

Case 2: A full core discharged to the SFP after 150 hours of decay time, plus fuel assemblies from 14 previous refueling discharges, with both SFP cooling trains operating, and

Case 3: A refueling load discharged (1/3 of a core) discharged to the SFP, then a full core offload 36 days later after 150 hours of decay time, plus 1/3 of a core with 400 days of decay, with both SFP cooling trains operating.

For BVPS Unit No. 2, Cases 1a and 3 addressed the "maximum normal" and "maximum abnormal" heat loads experienced by the SFP, respectively, as required by the Standard Review Plan.

Through this RAI and subsequent discussions with the NRC staff regarding the change in decay time from 150 hours to 100 hours, the cases to be analyzed have been redefined. The "normal" offload case is now to be interpreted to reflect the "planned" refueling practice of full core offloads, with consideration of the worst single failure to the SFP cooling systems. The "abnormal" offload case is now interpreted as the "unplanned" or emergency offload case where a full core offload is required 36 days after the last 1/3 of a core discharge has occurred, with no requirement to consider the worst single failure.

Both the planned and unplanned scenarios are to consider the heat load with all remaining spaces in the pool being filled with previously offloaded fuel assemblies. Additionally, the heat load re-analyses incorporated the fuel conditions from operations at 2918 megawatts thermal to address future operations at a planned large (9.4 percent) uprated power condition, a discharge rate of 6 fuel assemblies per hour, and the change in the decay time from 150 hours to 100 hours.

The FENOC re-analyses of SFP thermal-hydraulic performance were performed using the assumptions described above. The results for the maximum normal offload case are as follows:

Maximum "Planned" Offload Heat Load

BVPS Unit	Peak Spent Fuel Pool Temperature	Assumed Peak Component Cooling Water Temperature
Unit No. 1	170 °F	91.4 °F
Unit No. 2	170 °F	95.4 °F

The re-analyses of SFP thermal-hydraulic performance included results for other decay times greater than 100 hours. These analytical results are combined into graphs of decay time versus component cooling water (CCW) inlet temperature representing the conditions that will limit the SFP temperature to 170 °F for Unit Nos. 1 and 2, respectively. These graphs are presented in corresponding Attachments D-1 and D-2. Administrative controls will be established in accordance with the commitment stated in Attachment E.

NRC RAI Question 2

The single active failure assumed in Analysis Case 1a [the maximum normal offload case] should be the worst single active failure, including common cause failures.

FENOC Response

The FENOC reanalysis of SFP thermal-hydraulic performance was performed considering the worst single active failure. The worst single active failure, including consideration of common cause failures, for BVPS is considered to be the loss of one SFP cooling pump. It is important to note that the loss of one SFP cooling pump does not render the remaining train components unavailable. The heat exchanger from the train with the lost pump remains available for cooling through alignment to the operable train of SFP cooling.

NRC RAI Question 3

Your unplanned offload analysis should assume a decay heat load based on a full core offload plus refueling load that has decayed for 36 days plus heat load from a SFP with all other storage locations filled. In this case no single failure needs to be considered.

FENOC Response

The FENOC reanalysis of the "unplanned" offload cases were performed using the conditions as stated in NRC RAI Question 3 and discussed in response to RAI Question 1. The results for the unplanned offload cases are as follows:

Maximum "Unplanned" Offload Heat Load

BVPS Unit	Peak Pool Temperature	Assumed Peak CCW Temp.
Unit No. 1	168.7 °F	91.4 °F
Unit No. 2	172.7 °F	95.4 °F

NRC RAI Question 4

If your analysis shows that the spent fuel cooling systems cannot maintain spent fuel [pool] temperature below 150 °F under normal (planned) offload conditions, please submit an analysis that demonstrates that the SFP can withstand the higher temperature. This is based on the concrete code ACI-349-85 that states temperatures shall not exceed 150 °F for normal operation or any other long term periods of time.

FENOC Response

The analyses for Unit 1 and Unit 2 to calculate the pool water temperatures for normal (planned) offload conditions with no failures is the basis for the structural loading case incorporating the normal concrete temperature. These analyses have concluded that the maximum, bulk pool water temperatures for Units 1 and 2 are 155.7 °F and 159.2 °F, respectively. The duration of the period in which the bulk pool water temperature exceeds 150 °F is, approximately, 60 hours for Unit 1 and 100 hours for Unit 2.

During these intervals, the heated concrete surface adjacent to the pool water will experience a peak temperature equal to or less than the bulk pool water. The depth of the concrete subject to temperature in excess of 150 °F will be limited. The calculations for the spent fuel pool structures demonstrate that the reinforcement and concrete stresses are within the allowable limits for conditions including thermal loads corresponding to a pool temperature up to 175 °F. Temperatures in concrete are restricted to limit the loss in compressive strength resulting from the exposure to elevated temperatures. The structural calculations demonstrate that the maximum compressive stresses in concrete are less than 800 psi for loading combinations incorporating a concrete surface temperature up to 175 °F. These stresses are well below the design compressive strength of 3,000 psi. For normal structural loads, the effects of pool water temperatures exceeding 150 °F will not limit the concrete compressive stresses.

For abnormal refueling conditions and for normal refueling with a single system component failure, the structural reanalysis addressing the resulting temperature increases demonstrate that the SFPs can withstand higher temperature loads for the anticipated duration of elevated temperatures. The Unit Nos. 1 and 2 SFP structures have been evaluated for a pool temperature of 175°F. This temperature is assumed to be at the heated surface and to decrease linearly to the exterior surface. The interior portion of the pool structure is subject to this elevated temperature which is less than the 200 °F permitted by ACI 349-85 for local areas under normal (long-term) load conditions, although BVPS Unit Nos. 1 and 2 are not committed to use ACI 349-85. The results of the analyses show that the changes in the spent fuel pool concrete temperature loading due to power uprate and the described refueling scenarios result in concrete temperatures and reinforcement stresses that meet the structural acceptance criteria for abnormal (structural) conditions. The analyses calculating the Unit 1 and Unit 2 pool water temperatures for these abnormal structural conditions assume that the fuel remains in the SFP indefinitely. However, the offloaded fuel is not maintained in the SFP indefinitely. For the normal offload case, approximately 2/3 of the offloaded assemblies are returned to the reactor pressure vessel. The full core that would be offloaded in the abnormal case would return to the reactor pressure vessel once the initiating emergency condition is cleared or resolved. In both of these scenarios the heat load to the SFP would be reduced significantly.

ACI 349-85, Appendix A, Thermal Considerations, specifies a limitation of 150°F for concrete under normal operation or any other long term period except for local areas which are allowed to have increased temperatures not to exceed 200°F. Long-term temperature effects to the concrete fuel pool are associated with steady state conditions. Under accident or other short-term periods ACI 349-85 allows for increased concrete surface temperatures up to 350°F. Short-term temperature effects to the concrete fuel pool are associated with transient conditions. ACI 349-85 further stresses that judgement is required when evaluating the effects of accident (short-term) temperatures since they are dependent on the duration and location of the thermal transients, as well as the performance requirements for the structure.

NRC RAI Question 5

Your analysis should confirm that the SFP make-up source can provide make-up water equal to or greater than the boil-off rate and that make-up water can be provided within a sufficient time.

FENOC Response

The reanalysis shows that the highest evaporation rate occurs at boiling if all cooling is lost after abnormal refueling. For this case, in order to maintain pool level, the rate of replacement by makeup (at 100°F) is 101.2 gallons-per-minute (gpm) at BVPS Unit No. 1, and 96.2 gpm at BVPS Unit No. 2. As indicated in the analyses, the minimum times-to-boil (i.e., hours after peak pool temperature) for the abnormal off-load scenario are 2.33 hours and 2.58 hours for Unit Nos. 1 and 2, respectively.

For Unit No.1, provisions for makeup of spent fuel pool water are provided by connections from the Primary Grade Water supply, the Refueling Water Storage Tank cooling systems, an engine driven fire pump (via fire hose racks), and return of fuel building air conditioning condensate to the fuel pool.

For Unit No. 2, normal makeup water for the spent fuel pool is provided by the Primary Grade Water system. Borated makeup water may be supplied from the refueling water storage tank (RWST) through the fuel pool cleanup system. A backup supply of makeup can also be provided from the Fire Protection System, which has hose racks available in the fuel building. Makeup can also be supplied by return of fuel building air conditioning condensate to the fuel pool.

Procedures and equipment are available to facilitate operator response to conditions requiring the addition of inventory to the SFP and to facilitate restoration or alternate alignment SFP cooling system pumps at both Unit Nos. 1 and 2. Apart from the fuel building air conditioning condensate return, the various other water system sources that can be aligned to makeup to the SFP exceed the worst case maximum evaporation rates. The procedures and equipment available ensure that the actions required to establish the various makeup alignments can be performed in an expeditious manner and well within the worst case times-to-boil for each Unit.

SFP Makeup System Flow Rates

SFP Makeup Water Source	Flow Rate	Time Required to Place In-Service
Primary Grade Water		
Unit No. 1	>170 gpm	Within 30 minutes
Unit No. 2	168 gpm	Within 30 minutes
RWST Cooling Systems		
Unit No. 1	125 gpm	Within 30 minutes
Unit No. 2	75 gpm	Within 30 minutes
Engine Driven Fire Pump		
Unit No. 1 (3 hoses)	100 gpm each hose	Within 30 minutes
Unit No. 2 (2 hoses)	100 gpm each hose	Within 30 minutes
AC Condensate Return to SFP		
Unit No. 1	Insignificant	Within 30 minutes
Unit No. 2	Insignificant	Within 30 minutes
Service Water Systems		
Unit No. 1	Neglected	N/A
Unit No. 2	Neglected	N/A

ATTACHMENT D-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 281

Spent Fuel Pool Performance
Decay Time versus Component Cooling Water Temperature

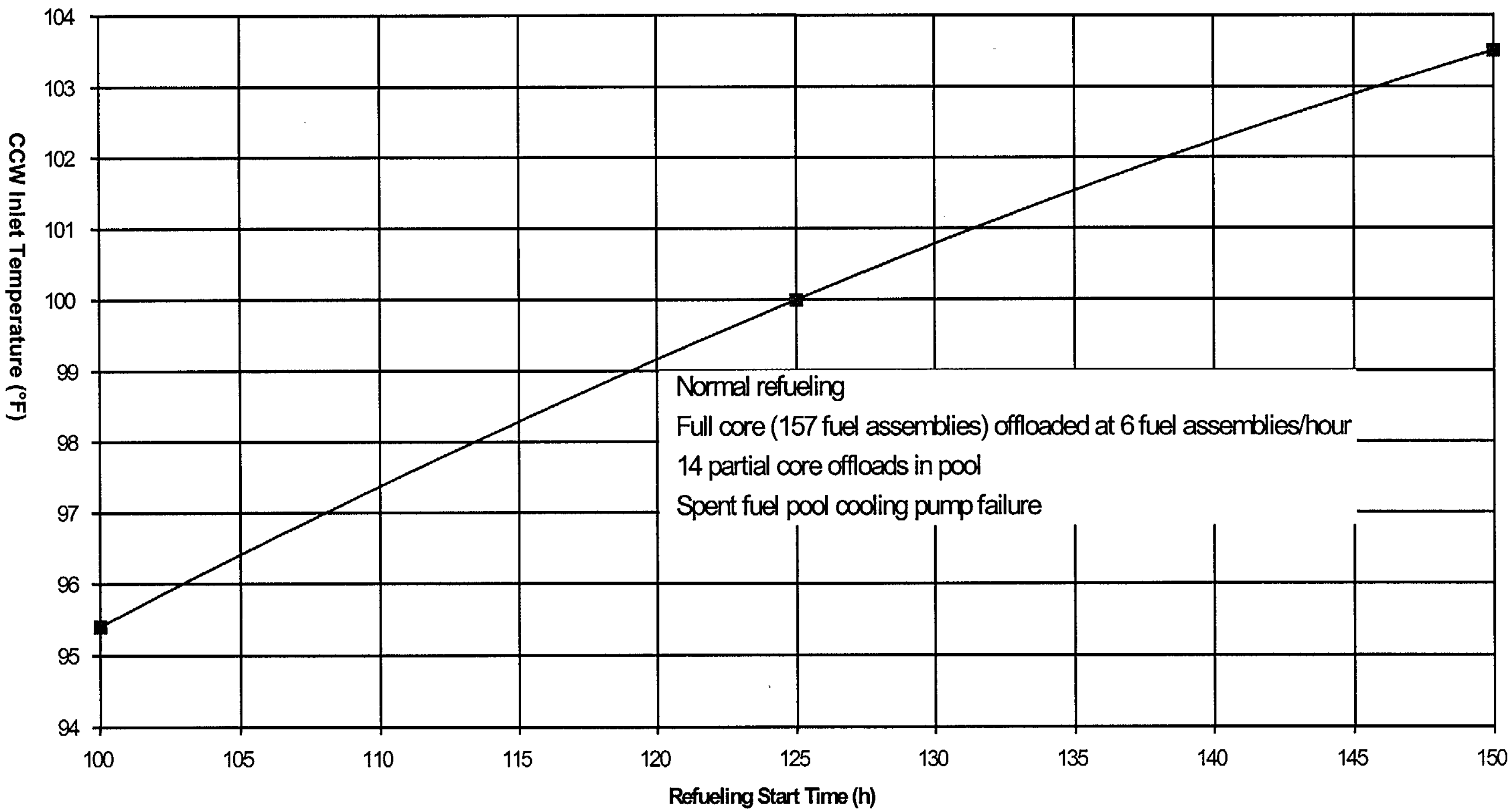
ATTACHMENT D-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 152

Spent Fuel Pool Performance
Decay Time versus Component Cooling Water Temperature

Beaver Valley Unit 2

CCW Inlet Temperature vs. Refueling Start Time



ATTACHMENT E

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 281 and 152

Commitment Summary

Commitment List

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for Beaver Valley Power Station (BVPS) Unit 2 in this document. Any other actions discussed in the submittal represent intended or planned actions by Beaver Valley. They are described only as information and are not regulatory commitments. Please notify Mr. Thomas S. Cosgrove, Manager, Regulatory Affairs, at Beaver Valley on (724) 682-5203 of any questions regarding this document or associated regulatory commitments.

Commitment	**Due Date**
Administrative controls will be established to control decay times prior to the movement of irradiated fuel assemblies from the reactor core based on the "Decay Time versus Component Cooling Water" curves provided in Attachments D-1 and D-2.	Prior to refueling outages 1R15 and 2R10 for Units 1 and 2, respectively.

ATTACHMENT F-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 281

Proposed Technical Specification Bases Pages
(For Information Only)

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless ~~150~~100 hours of decay has occurred~~, which is conservative with respect to the assumptions used in the accident analyses~~. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

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

The requirements on containment penetration closure and operability of the containment purge and exhaust system HEPA filters and charcoal adsorbers ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere within 10 CFR 50.67 limits. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from the number of fuel rods assumed to be ruptured in the FHA analysis 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.

3/4.9.5 COMMUNICATIONS

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

3/4.9.6 MANIPULATOR CRANE OPERABILITY

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

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

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, 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 LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the containment to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless ~~150~~100 hours of decay

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM (Continued)

has occurred~~, which is conservative with respect to the assumptions used in the accident analyses~~. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

THE OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The integrity of the containment penetrations of this system may be required to restrict the release of radioactive material from the containment atmosphere to acceptable levels which are less than those listed in 10 CFR 50.67.

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

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

3/4.9.12 FUEL BUILDING VENTILATION SYSTEM

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the fuel building to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless ~~150~~100 hours of decay has occurred~~, which is conservative with respect to the assumptions used in the accident analyses~~. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

The limitations on the storage pool ventilation system ensure that all radioactive material released, as a result of a fuel handling accident (FHA) within the fuel building involving recently irradiated fuel, will be filtered through the HEPA filters and charcoal adsorber

ATTACHMENT F-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 152

Proposed Technical Specification Bases Pages
(For Information Only)

3/4.7.5 ULTIMATE HEAT SINK (Continued)

exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants."

3/4.7.6 FLOOD PROTECTION

The limitation on flood level ensures that facility operation will be terminated in the event of flood conditions. The limit of elevation 695 Mean Sea Level was selected on an arbitrary basis as an appropriate flood level at which to terminate further operation and initiate flood protection measures for safety related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM

This LCO is applicable during MODES 1, 2, 3 and 4. This LCO is also applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) for which the requirements of this Specification may be required to limit radiation exposure to personnel occupying the control room. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposure, to personnel occupying the control room, that is within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit personnel exposure. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless ~~150~~100 hours of decay has occurred~~, which is conservative with respect to the assumptions used in the accident analyses~~. Therefore, this specification will not be applicable, during fuel movement, unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

\The OPERABILITY of the control room emergency air cleanup and pressurization system ensures that the control room will remain habitable with respect to potential radiation hazards for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent, or 5 rem TEDE, as applicable. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50 or 10 CFR 50.67, as applicable.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the containment to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless ~~150~~100 hours of decay has occurred~~, which is conservative with respect to the assumptions used in the accident analyses~~. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

The requirements on containment penetration closure limit leakage of radioactive material within containment to the environment may be required to ensure compliance with 10 CFR 50.67 limits. The requirements on operation of the SLCRS ensure that radioactive material released through open containment penetrations, as the result of a fuel handling accident (FHA) within containment involving recently irradiated fuel, will be filtered through HEPA filters and charcoal absorbers prior to discharge to the atmosphere. These requirements are sufficient to restrict radioactive material release from the number of fuel rods assumed to be ruptured in the FHA analysis based upon the lack of containment pressurization potential while moving fuel assemblies within containment.

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

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the containment to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless ~~150~~100 hours of decay has occurred~~, which is conservative with respect to the assumptions used in the accident analyses~~. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

THE OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The integrity of the containment penetrations of this system may be required to meet 10 CFR 50.67 requirements in the event of a fuel handling accident inside containment involving recently irradiated fuel. 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.

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

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

3/4.9.12 FUEL BUILDING VENTILATION SYSTEM

The LCO is applicable during movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies because there is a potential for the limiting fuel handling accident (FHA) to occur. Therefore, the requirements of this Specification may be required to limit leakage of radioactive material within the fuel building to the environment. A FHA which does not involve recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 100 hours) will result in radiation exposures that are within the guideline values specified in 10 CFR 50.67 without any reliance on the requirements of this Specification to limit leakage to the environment. The 100 hour limit is based on the current radiological analysis for a FHA which assumes a decay time of 100 hours. LCO 3.9.3 prohibits irradiated fuel movement unless ~~150~~100 hours of decay has occurred~~, which is conservative with respect to the assumptions used in the accident analyses~~. Therefore, this specification will not be applicable unless the decay time in Specification 3.9.3 and the time assumed in the radiological analysis for a FHA are reduced to below 100 hours.

The limitations on the storage pool ventilation system ensure that all radioactive material released, as a result of a fuel handling accident (FHA) within the fuel building involving recently irradiated fuel, will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The spent fuel pool area ventilation system is non-safety related and only recirculates air through the fuel building. The fuel building portion of the SLCRS is safety related and continuously filters the fuel building exhaust air. This maintains a negative pressure in the fuel building.

3/4.9.13 (This Specification is not used.)