August 30, 2001

Mr. L. W. Myers
Senior Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: REVISED FUEL HANDLING ACCIDENT SAFETY ANALYSIS AND REQUIREMENTS FOR HANDLING IRRADIATED FUEL ASSEMBLIES IN THE REACTOR CONTAINMENT AND IN THE FUEL BUILDING (TAC NOS. MB1492 AND MB1493)

Dear Mr. Myers:

The Commission has issued the enclosed Amendment No. 241to Facility Operating License No. DPR-66 and Amendment No. 121 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2). These amendments consist of changes to the Technical Specifications (TSs) and Updated Final Safety Analysis Report (UFSAR) in response to your application dated March 19, 2001, as supplemented by letters dated July 6, August 8, and August 23, 2001.

These amendments revise the BVPS-1 and 2 UFSAR radiological dose consequence analyses of a postulated fuel handling accident (FHA). Consistent with these revised analyses, the BVPS-1 and 2 TSs associated with requirements for handling irradiated fuel assemblies in the reactor containment and fuel building are revised. The proposed amendment also revises the TSs associated with ensuring that safety analyses assumptions are met for a postulated FHA. The revised FHA analyses constitute a selective implementation of an alternative radiological source term in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67, “Accident Source Term.”

A copy of our safety evaluation is also enclosed. The Notice of Issuance will be forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures: 1. Amendment No. 241 to DPR-66
2. Amendment No. 121 to NPF-73
3. Safety Evaluation
4. Notice of Issuance

cc w/encls: See next page

A copy of our safety evaluation is also enclosed. The Notice of Issuance will be forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

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3. Safety Evaluation
4. Notice of Issuance

cc w/encls: See next page

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EAdensam	OGC	GHubbard	CHolden
GHill (4)	WBeckner	BThomas	PDI-1,(A)SC

ACCESSION NO. ML012330496

*No significant changes made to SE input
**See previous concurrence

OFFICE	PDI-1/PM	PDI-2/LA	SPSB/SC	EEIBA/SC	EEIBB/SC
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PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 241
License No. DPR-66

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee) dated March 19, 2001, as supplemented by letters dated July 6, August 8, and August 23, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, changes to the Updated Final Safety Analysis Report (UFSAR) to reflect the revisions made to the radiological dose consequence analysis of a postulated fuel handling accident as described in the attached safety evaluation and as set forth in the application for amendment dated March 19, 2001, as supplemented by letters dated July 6, August 8, and August 23, 2001, are authorized.

3. In addition, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 241, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. This license amendment is effective as of the date of its issuance. The Technical Specification changes shall be implemented within 60 days. The UFSAR changes shall be implemented by the next update as required by 10 CFR 50.71(e). Implementation of the amendment requires incorporation in the UFSAR of the changes made to the description of the facility as described in the licensee's application dated March 19, 2001, as supplemented by letters dated July 6, August 8, and August 23, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Timothy G. Colburn, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 30, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 241

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
VIII	VIII
IX	IX
XIII	XIII
3/4 3-34	3/4 3-34
3/4 3-35	3/4 3-35
3/4 3-35a	3/4 3-35a
3/4 3-36	3/4 3-36
3/4 6-18	3/4 6-18
3/4 7-16	3/4 7-16
3/4 7-16a	3/4 7-16a
3/4 8-5	3/4 8-5
3/4 8-7	3/4 8-7
3/4 8-10	3/4 8-10
3/4 9-3	3/4 9-3
3/4 9-4	3/4 9-4
3/4 9-9	3/4 9-9
3/4 9-10	3/4 9-10
3/4 9-11	3/4 9-11
3/4 9-12	3/4 9-12
3/4 9-13	3/4 9-13
3/4 9-16	-
3/4 9-17	-
B 3/4 3-2	B 3/4 3-2
B 3/4 7-5	B 3/4 7-5
B 3/4 7-6	B 3/4 7-6
B 3/4 7-6a	B 3/4 7-6a
B 3/4 8-1	B 3/4 8-1
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3
B 3/4 9-4	B 3/4 9-4
B 3/4 9-5	B 3/4 9-5
-	B 3/4 9-6

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
License No. NPF-73

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee) dated March 19, 2001, as supplemented by letters dated July 6, August 8, and August 23, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, changes to the Updated Final Safety Analysis Report (UFSAR) to reflect the revisions made to the radiological dose consequence analysis of a postulated fuel handling accident as described in the attached safety evaluation and as set forth in the application for amendment dated March 19, 2001, as supplemented by letters dated July 6, August 8, and August 23, 2001, are authorized.

3. In addition, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 121, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

4. This license amendment is effective as of the date of its issuance. The Technical Specification changes shall be implemented within 120 days. The UFSAR changes shall be implemented by the next update as required by 10 CFR 50.71(e). Implementation of the amendment requires incorporation in the UFSAR of the changes made to the description of the facility as described in the licensee's application dated March 19, 2001, as supplemented by letters dated July 6, August 8, and August 23, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Timothy G. Colburn, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 30, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 121

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following pages of Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
IX	IX
XII	XII
XIII	XIII
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 3-42	3/4 3-42
3/4 3-43	3/4 3-43
3/4 3-44	3/4 3-44
3/4 6-16	3/4 6-16
3/4 7-15	3/4 7-15
3/4 7-16	3/4 7-16
3/4 7-17	3/4 7-17
-	3/4 7-17a
3/4 8-6	3/4 8-6
3/4 8-8	3/4 8-8
3/4 8-12	3/4 8-12
3/4 9-3	3/4 9-3
3/4 9-4a	3/4 9-4a
3/4 9-4b	3/4 9-4b
3/4 9-10	3/4 9-10
3/4 9-11	3/4 9-11
3/4 9-12	3/4 9-12
3/4 9-13	3/4 9-13
3/4 9-14	3/4 9-14
B 3/4 3-10	B 3/4 3-10
B 3/4 7-4	B 3/4 7-4
B 3/4 7-5	B 3/4 7-5
-	B 3/4 7-5a
B 3/4 8-1	B 3/4 8-1
B 3/4 8-2	B 3/4 8-2
B 3/4 8-3	B 3/4 8-3
B 3/4 9-1	B 3/4 9-1
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3
B 3/4 9-4	B 3/4 9-4
B 3/4 9-5	B 3/4 9-5
B 3/4 9-6	B 3/4 9-6
B 3/4 9-7	B 3/4 9-7
-	B 3/4 9-8
-	B 3/4 9-9

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 241 AND 121 TO FACILITY OPERATING

LICENSE NOS. DPR-66 AND NPF-73

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By letter dated March 19, 2001 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML010810433), as supplemented by letters dated July 6 (ADAMS Accession No. ML011980423), August 8 (ADAMS Accession No. ML012260302), and August 23, 2001 (ADAMS Accession No. ML012420089), the FirstEnergy Nuclear Operating Company (FENOC; the licensee) submitted a request for changes to the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2), Updated Final Safety Analysis Report (UFSAR) and Technical Specifications (TSs).

The licensee's submittals proposed revisions to the BVPS-1 and 2 UFSAR radiological dose consequence analyses of a postulated fuel handling accident (FHA). The information supporting these revisions demonstrates that the consequences of an FHA, once the fuel has undergone radioactive decay for 100 hours, would result in calculated radiation exposures within the guidelines of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.67, "Accident Source Term." Consistent with the assumptions and description of the revised FHA analysis, the licensee also proposed revisions to the BVPS-1 and 2 TSs associated with the requirements for handling irradiated fuel assemblies in the reactor containment and fuel building. The proposed amendment would also revise the TS requirements associated with ensuring that safety analysis assumptions are met for a postulated FHA. The term "recently irradiated" fuel would be defined in the applicable TS Bases as "fuel that has occupied part of a critical reactor core within the previous 100 hours." The purpose of the addition of the term "recently irradiated" throughout the TSs is to establish a point where operability of those systems typically used to mitigate the consequences of an FHA is no longer required to meet the radiation exposure limits of 10 CFR 50.67. This amendment revises the TSs to eliminate

TS controls over the integrity of the fuel building and the reactor containment building and the operability of the associated building's ventilation/filtration systems after the decay period of 100 hours.

The FHA radiological analyses that were performed by the licensee to support the proposed TS changes use an alternative radiological source term (AST). This submittal constitutes a selective implementation of an AST for analysis of the FHA only.

The July 6, 2001, letter withdrew the portion of the original request related to decreasing the decay time specified in TS 3.9.3, "Decay Time." Consequently, Limiting Condition of Operation (LCO) 3.9.3 will continue to require the reactor to be subcritical for at least 150 hours prior to movement of fuel in the reactor pressure vessel.

The July 6, August 8, and August 23, 2001, letters provided clarifying information that did not expand the scope of the original *Federal Register* notice.

2.0 EVALUATION

2.1 Changes to Design-Basis FHA Analyses

The Nuclear Regulatory Commission (NRC) staff reviewed the licensee's revised radiological analysis of the design-basis FHA, which assumed no credit for building integrity requirements or ventilation and filtration systems. Additionally, the licensee proposed to selectively implement an alternative source term in accordance with 10 CFR 50.67 and has followed guidance put forth in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000." The NRC staff used guidance in NUREG 0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 15.0.1, to aid in its review.

For BVPS-1 and 2, the licensee performed an analysis of the radiological consequences of a design-basis FHA. The doses to personnel within the control room and offsite doses to a member of the public at the Emergency Area Boundary (EAB) and Low Population Zone (LPZ) were calculated. The licensee's analysis for each unit assumed the FHA accident occurs 100 hours after reactor shutdown with the worst-case conditions that may be encountered in either the fuel handling building or the reactor containment building. The analyses assumed release over a 2-hour period of all the radioactive gases from the maximum calculated number of damaged fuel assembly rods. This radioactivity was assumed to be directly released to the environment with no credit for filtration or holdup in the surrounding building. The NRC staff finds that the assumptions, inputs, and methodologies used in the analyses of the FHA for BVPS-1 and 2 generally follow the guidance in RG 1.183. Some specific assumptions warranted closer inspection by the NRC staff and are discussed below.

2.1.1 Use of Atmospheric Dispersion Factors

The licensee uses the current UFSAR atmospheric dispersion factors or chi-over-Q's (X/Qs) for calculating the doses both offsite and in the control room. The licensee stated that the X/Q for the most restrictive release point for either building was conservatively used in its analyses. These X/Qs were not calculated assuming the likely release points if building integrity is not required while moving fuel, such as an open containment equipment hatch. However, the NRC

staff finds that the current UFSAR X/Qs are applicable under the proposed TS conditions based on the following discussion.

The NRC staff reviewed previously docketed material on calculation of the licensee's current licensing basis offsite X/Qs, Beaver Valley calculation ERS-SFL-96-021, Rev. 0, "RG 1.145 Short-Term Accident X/Q Values for EAB and LPZ, Unit 1 and Unit 2, Based on 1986-1995 Observations." These X/Qs were approved by License Amendment No. 205 to Facility Operating License (FOL) DPR-66, dated September 10, 1997 (ADAMS Accession No. ML00376854). Revision 2 of this calculation is referenced in the current submittal, and the X/Q values used in the current submittal are the same as those reported in Revision 0 and the UFSAR for BVPS-1 and 2. Because the BVPS-1 and 2 current offsite X/Qs were calculated using parameters that bound those that would be used to calculate X/Qs for a release from the open equipment hatch or other likely release points at that unit, the NRC staff finds that the current licensing basis offsite X/Qs are bounding and are acceptable for use in calculating the radiological consequences of an FHA under conditions of the proposed changes to the TSs.

The current control room X/Qs were approved by License Amendment No. 57 to FOL NPF-73, dated September 28, 1993 (ADAMS Accession No. ML003772518). The NRC staff finds that the current licensing basis X/Qs are conservative and appropriate for use in the revised FHA analyses, and that the current "primary auxiliary building to control room" X/Qs are bounding for a release from the open equipment hatch or other likely release points.

The NRC staff finds that the use of the current control room and offsite X/Qs is appropriate and acceptable for use in the revised design-basis FHA dose consequence analyses.

2.1.2 Evaluation of Fuel Assembly Drop Accident Analysis

SRP, Section 9.1.2, "Spent Fuel Storage," provides criteria for the NRC staff to review the design and performance of the spent fuel storage and fuel handling systems, respectively. Accordingly, the NRC staff's review focuses on (1) the maintenance of the safety function of the spent fuel pool and storage racks, (2) the assurance that the spent fuel assemblies are in a safe and critical array during all credible storage conditions, and (3) the provision of a safe means of loading the assemblies into shipping casks.

SRP, Section 9.1.4, "Light Load Handling System," also provides criteria for the NRC staff to review the design and performance of the light load handling system used to move the spent fuel assembly into the shipping cask. NRC staff application of the criteria is focused on, among other things, the adequacy of the methods and equipment for transferring stored fuel to and from the spent fuel pool.

SRP, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," provides criteria for the NRC staff to review the adequacy of the system design features and plant procedures provided for the mitigation of radiological consequences of accidents that involve damage to spent fuel. The accidents are limited to the drop of a single spent fuel assembly and its handling tool or of a heavy object onto other spent fuel assemblies. NRC staff application of these criteria is focused on, among other things, the values of the plant parameters for use in calculating the radiological consequences of a design-basis FHA. NRC staff review of the FHA is focused on the adequacy of the analysis to address all the significant postulated fuel drop

events that could occur, the number of fuel rods assumed to be damaged in an FHA, and the potential impact on maintaining the containment in a non-isolated configuration.

RG 1.13, "Spent Fuel Storage Facility Design Basis," provides methods that are acceptable to the NRC staff for implementing General Design Criterion (GDC) 61 of Appendix A to 10 CFR Part 50 which requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions.

RG 1.183 provides assumptions for evaluating the radiological consequences of an FHA, including parameters such as the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, the compression, torsion, and shear stresses on the irradiated fuel rods, and damage to adjacent or impacted fuel assemblies.

FHA (Bounding) Analysis

Consistent with the assumptions and description of the revised FHA analysis, the licensee proposes to revise the BVPS-1 and 2 TSs associated with the requirements for handling irradiated fuel assemblies in the reactor containment and fuel building. Additionally, the licensee proposes to revise the TSs associated with ensuring that safety analysis assumptions are met for a postulated FHA. The term "recently irradiated" fuel would be defined in the applicable TS Bases as "fuel that has occupied part of a critical reactor core within the previous 100 hours." The purpose of the addition of the term "recently" throughout the TSs is to establish a point where operability of those systems typically used to mitigate the consequences of an FHA is no longer required to meet the radiation exposure limits of 10 CFR 50.67, i.e., this amendment would revise the TSs to eliminate TS controls over the integrity of the fuel building and the reactor containment building and the operability of the associated building's ventilation/filtration systems after the decay period of 100 hours.

To support the proposed revisions to the TSs, the revised design-bases postulated FHA analysis evaluates a potential drop of a fuel assembly that could result in the most severe consequences, including the potential for the release of significant amounts of radioactive material to the environment. The FHA is defined as the dropping of one spent fuel assembly onto other fuel assemblies in the spent fuel storage area. The dropped fuel assembly is a Westinghouse 17 x 17 (Optimized Fuel Assembly [OFA]). It is considered an OFA because it contains fuel rods of the smallest diameter of pressurized water reactor (PWR) spent fuel assemblies that are most vulnerable to drop accidents.

The current design-basis FHA postulates a dropped fuel assembly in the spent fuel storage area of the fuel building and in the containment building. It involves a dropped assembly (264 rods) and the impacted assembly (34 rods) of a total of 298 rods for BVPS-1, and 2.34 assemblies (617 rods) for BVPS-2. Consequently, the current design-basis FHA analysis assumes failure of 298 rods and 617 rods for BVPS-1 and BVPS-2, respectively. These numbers were based on the conservative assumptions provided in RG 1.183 which recommends that the FHA analysis present a conservative analysis that considers the most limiting case.

The total weight of a spent fuel assembly and its handling tool (2450 pounds) is defined as a heavy load in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." In the revised FHA analysis, the licensee is conservatively increasing the total weight

from 2450 pounds to 2500 pounds. The drop height is to be increased from approximately 23 feet to 30 feet.

The revised design-basis FHA dose consequence analysis assumes that irradiated fuel decays for at least 100 hours after reactor shutdown and before the first fuel assembly is moved.

Four fuel drop cases were postulated to occur in the reactor core. These four fuel drop cases were analyzed in the Holtec International Report No. HI-992343, “Evaluations of Spent Fuel Assembly Drop Accidents in the Beaver Valley Power Station Reactor Core,” (which was submitted as Attachments D-2 [proprietary] and D-3 [nonproprietary] to FENOC’s March 19, 2001, letter). The four fuel drop cases are bounding for all possible drop events that could result in the potential release of significant amounts of radioactive material. These fuel drop cases and the results are based on the use of a finite element analysis model (the LS-DYNA) of the actual configuration of the Westinghouse 17 x 17 OFA fuel assembly. The LS-DYNA, Version 950, is a commercial computer numerical code that was developed by the Livermore Software Technology Corporation and quality assurance validated by Holtec International. The cases of dropped and impacted fuel assemblies were modeled using the LS-DYNA. The cases of dropped fuel assemblies and the results are as follows.

Case No. 1 involves a spent fuel assembly dropped vertically 30 feet through the water impacting the rigid reactor core support with all the end fitting pedestals, resulting in 74 ruptured fuel rods.

Case No. 2 involves a spent fuel assembly dropped vertically approximately 16.7 feet in which the end fittings of the dropped assembly center-strikes the top fittings of the targeted fuel assembly stored in the reactor core. This results in 137 ruptured fuel rods with a majority of the ruptured fuel rods (119) in the targeted fuel assembly and the remaining ruptured fuel rods (18) in the dropped fuel assembly.

Case No. 3 involves a spent fuel assembly dropped vertically approximately 16.7 feet that hits the edge of the targeted fuel assembly stored in the reactor core, resulting in 46 ruptured rods.

Case No. 4 involves a tipped-over spent fuel assembly that rotates and angularly impacts the side of a standing fuel assembly which then falls over and impacts the reactor core support. The results of the analysis of this accident indicate that no fuel rods would be ruptured in either fuel assembly and there would be no damage to the fuel assemblies themselves.

Therefore, the analyses show that the maximum expected damage for the accident occurring in the reactor containment is 137 rods (Case No. 2).

The NRC staff’s review of the Holtec report included the modeling methodology of the fuel assembly and the results of the impact analysis of postulated fuel assembly drop accidents. The licensee, in Section 6.2 of the report, stated that the material properties of structural components in the fuel assembly could deviate from the nominal value significantly when deformation is applied in an impulsive manner or the material is irradiated for a long period of time. In its August 23, 2001, letter, the licensee further stated that the material properties used in the analysis are the best estimates produced from the test data reported in Reference 9.5 of the report. The referenced tests were performed on irradiated fuel at strain rates expected to

be present in the reactor and resulted in a conclusion that the effects of material irradiation and strain rate had been included in the analysis.

Also in its letter dated August 23, 2001, regarding sensitivity of the analysis results to the variation in material properties, the licensee stated that it performed additional analyses for the dropped fuel assembly impacting onto a target fuel assembly assuming a 10 percent reduction in cladding failure stress value used in the original analysis. The results with reduced cladding failure stress indicate that a reduction in failure stress of up to 10 percent may result in up to a 26 percent increase (36 rods) in the number of reported rod failures. The licensee stated that, based on the originally reported 137 fuel rod failures, it had calculated the most restrictive dose of 2.2 rem for the control room compared to the 10 CFR 50.67 limit of 5 rem. The licensee stated that the dose is directly proportional to the assumed number of ruptured fuel rods and a maximum of 311 fuel rods could be ruptured before exceeding the 5 rem control room dose limit. Based on its review, the NRC staff concludes that there is adequate design margin for variation in the material properties of the fuel assembly structural components.

Based on its review of the Holtec International report and the licensee's August 23, 2001, letter, the NRC staff finds that the licensee's modeling methodology and the results of the spent fuel assembly drop accident analyses are reasonable and acceptable.

FHA Analysis - Fuel Building

The above discussed bounding FHA analysis envelopes the FHA that could occur in the fuel building. In addition, the potential release of radioactivity from the pool during an FHA in the fuel building would result in a release in the fuel building atmosphere that would be filtered by the supplemental leak collection and release system (SLCRS). The SLCRS processes radioactive releases following a design-basis accident (DBA) in primary containment or releases due to an FHA in the fuel building. Subsequent to a high-high radiation alarm, a release in the fuel building is collected and filtered in order to remove the iodine particulates prior to discharge to the atmosphere through a ventilation vent at an elevated release point.

The NRC staff finds that the proposed revision to the weight of a PWR spent fuel assembly from 2450 pounds to 2500 pounds is a more conservative weight for assessing the potential impact of a dropped spent fuel assembly and its handling tool (a heavy load per NUREG-0612). The NRC staff finds this change acceptable.

The NRC staff finds that the four FHA accident scenarios bound the potential accidents in the fuel and containment buildings. Additionally, the NRC staff finds that the changes to the spent fuel assembly drop accident analyses are more conservative or result from better-defined fuel drop scenarios and the use of more accurate numerical analyses. Therefore, the NRC staff finds that the revised spent fuel assembly drop accident analyses are acceptable for the BVPS-1 and 2 revised design-basis FHA dose consequence analyses (with Case No. 2 being the most limiting with a maximum of 137 ruptured rods).

2.1.3 Acceptability of Licensee Analyses

The NRC staff has reviewed the licensee's analyses as submitted and has determined that the assumptions, inputs, and methodology are consistent with regulatory guidance and are

acceptable. The NRC staff also performed confirmatory calculations using the licensee's stated assumptions and confirmed the licensee's results. For BVPS-1 and 2, the calculated results of the FHA radiological consequence analyses are within the total effective dose equivalent (TEDE) acceptance criteria given in RG 1.183, which are 6.3 rem TEDE for the EAB and LPZ and 5 rem TEDE for the control room.

The NRC staff has reviewed the AST selective implementation proposed by FENOC for BVPS-1 and 2. The NRC staff reviewed the assumptions, inputs, and methodology used by the licensee to assess the radiological impacts in the context of the AST. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative guidance of RG 1.183. The NRC staff compared the doses estimated by the licensee to the applicable acceptance criteria and the results estimated by the NRC staff in its confirmatory calculations. Furthermore, The NRC staff finds with reasonable assurance that the licensee's calculated estimates of the TEDE resulting from a postulated design-basis FHA comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183. Consequently, the NRC staff finds the proposed AST implementation acceptable.

With this approval, the selected characteristics of the AST and the TEDE criteria become the design basis for the FHA either in the fuel handling building or in the reactor containment building at BVPS-1 and 2. The selected characteristics of the AST and the TEDE criteria may not be extended to other aspects of the plant design or operation without prior NRC review under 10 CFR 50.67. All future radiological analyses performed to demonstrate compliance with the regulatory requirements should address the selected characteristics of the AST and the TEDE criteria as described in the facility design basis.

2.2 Revisions Common to the BVPS-1 and 2 TSs

2.2.1 Adding the term "recently irradiated" fuel to various TS Sections

The term "recently" is added to applicability and requirements preceding the word "irradiated" in the proposed changes for several TSs. The term "recently" when used in this context represents the decay period for the reduction of radionuclide inventory available for release in the event of an FHA. The licensee has determined that the calculated dose consequences resulting from a design-basis FHA will not exceed 10 CFR 50.67 dose limitations after a decay period of 100 hours. This change is consistent with NRC-approved Technical Specification Traveler Form 51 (TSTF-51), Revision 2, to NUREG-1431, Revision 1, "Standard Technical Specifications - Westinghouse Plants," which was approved on October 15, 1999. The NRC staff finds that the proposed change clarifies the existing requirement and is consistent with the revised radiological dose consequence analysis of a postulated FHA. Therefore, the proposed change to the applicability and action requirements in the following TSs and their Bases is acceptable:

1. For BVPS-1: 3.9.4, 3.9.9, and 3.9.12, and

2. For BVPS-2: 3.7.7, 3.9.4, 3.9.9, and 3.9.12.

In addition, and in accordance with the Reviewers Note in TSTF-51, the licensee committed to the following guidelines for the assessment systems removed from service during movement of irradiated fuel assemblies and movement of fuel assemblies over irradiated fuel assemblies:

> "During fuel handling/core alterations, ventilation system and radiation monitor availability (as defined in NUMARC 91-06) should be assessed, with respect to filtration and monitoring of releases from the fuel. Following shutdown, radioactivity in the RCS decays fairly rapidly. The basis of the Technical Specification operability amendment is the reduction in doses due to such decay. The goal of maintaining ventilation system and radiation monitor availability is to reduce doses even further below that provided by the natural decay, and to avoid unmonitored releases. A single normal or contingency method to promptly close primary or secondary containment penetrations should be developed. Such prompt methods need not completely block the penetration or be capable of resisting pressure. The purpose is to enable ventilation systems to draw the release from a postulated fuel handling accident in the proper direction such that it can be treated and monitored."

In Attachment E of its March 19, 2001, letter, the licensee provided its draft containment/fuel building closure controls. The areas addressed are similar to those previously approved by the NRC. The NRC staff reviewed the BVPS-1 and 2 draft containment/fuel building closure controls and conclude they are reasonable and adequate and, consequently, acceptable.

2.2.2 Revisions to TS 3/4.3, "Instrumentation" include:

4. Table 3.3-6, "Radiation Monitoring Instrumentation," Item 1.c. Change the Applicable Modes from 1, 2, 3, 4, 5$^{(4)}$ and 6$^{(4)}$ to 1, 2, 3, 4, and (4).

5. Table 4.3-3, "Radiation Monitoring Instrumentation Surveillance Requirements," Item 1.c. In the Modes in Which Surveillance Required column, change from 1, 2, 3, 4, 5##, 6## to 1, 2, 3, 4, and ##.

In its submittal, the licensee stated that during Modes 5 and 6, while fuel handling is not being performed, these monitors will not be required to be operable. This is because, during these periods, the potential for radioactive releases is minimized and operator action is sufficient to ensure that the post-accident offsite doses are maintained within the regulatory limits. The addition of the phrase, "During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies," is necessary based on the results of the fuel drop analysis as discussed in Section 2.1 of this safety evaluation (SE). The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod rupture. Additionally, the movement of a non-irradiated fuel assembly could result in a radiological release that might require the use of the control room habitability systems if this assembly were to strike an irradiated fuel assembly. The proposed Modes will continue to require these monitors to be operable when there is a potential for an FHA, that is, "during movement of recently irradiated fuel assemblies (within the containment) and during the movement of fuel assemblies over recently irradiated fuel assemblies (within the containment)." Therefore, the proposed change to the applicable modes is acceptable to the NRC staff. Since surveillance is not needed for any equipment that is not required to be operable, the proposed change in the

mode in which surveillance is required is consistent with the operability requirement for these monitors and, therefore, is acceptable to the NRC staff.

6. Table 3.3-6, “Radiation Monitoring Instrumentation,” Unit 1, Item 2.b, “Process Monitors, Fuel Storage Building Gross Activity (RMVS-103 A & B),” and Unit 2, Item 2.b.i, “Process Monitors, Fuel Building Vent Gaseous Activity (Xe -133) (2RMF - RQ301B),” and Item 2.b.ii, “Particulate (I - 131) (2RMF - RQ301A).” Delete the entire statement of requirements and replace with the word “Deleted.”

7. Table 4.3-3, “Radiation Monitoring Instrumentation Surveillance Requirements,” Item 2.b. Delete the entire statement of surveillance requirements (SRs) and replace with the word “Deleted.”

The proposed deletion of the fuel building process radiation monitors RM-1VS-103A&B for BVPS-1 and 2RMF-RQ301A&B for BVPS-2 TS requirements is justified since TS 3.9.12 will ensure that the fuel building exhaust flow is aligned to the SLCRS filter banks during plant evolutions when an FHA could occur and the filtration of the fuel building atmosphere may be required to meet the limits specified in 10 CFR 50.67. TS 3.9.12 requires that the fuel building exhaust flow be aligned to the SLCRS filter banks during fuel movement within the fuel building. The automatic action of the BVPS-1 fuel building process monitors to initiate filtration of the fuel building exhaust flow is, therefore, not required. The effluent monitors requirements contained in the BVPS-1 and 2 Offsite Dose Calculation Manual (ODCM) for the SLCRS elevated release point will not be affected by this proposed deletion of the fuel building monitors. The capability to monitor this SLCRS radioactivity release point to the environment will be maintained per the ODCM requirements. The TS requirements for the fuel pool storage area monitor RM-207 for BVPS-1 and 2RMF-RQ202 for BVPS-2 will also not be affected by the proposed change.

The improved standard TSs (ISTS) provide guidance as to the types and bases for the radiation monitors required by 10 CFR 50.36 to remain within the TSs. The radiation monitors retained in ISTS perform one of the following functional requirements that have been determined to meet the NRC’s Final Policy Statement on TS improvements (58 FR 39132) which provides criteria for requirements that should be included in the TSs. The radiation monitors required to be retained in the TSs are:

A. Assumed in the safety analysis to provide automatic initiation of a system or component that is part of the primary success path and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier (10 CFR 50.36, Criterion 3),

B. Assumed in the safety analysis to provide an indication or alarm that is relied on by operators to initiate manual action that is part of the primary success path and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier (10 CFR 50.36, Criterion 3),

C. Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary (10 CFR 50.36, Criterion 1), or

D. Instrumentation that monitors RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Type A or Category I variables. RG 1.97 Type A variables are those indications that provide the primary information required for the control room operators to take specific manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs (10 CFR 50.36, Criterion 3). In general RG 1.97 Category I (non-Type A) variables provide indications intended to assist operators in minimizing the consequences of accidents. As such, Category I instrumentation has been identified as important for reducing risk to the public (10 CFR 50.36, Criterion 4).

Based on the information provided by the licensee, the NRC staff finds that the fuel building process radiation monitors do not:

A. Provide an automatic initiation function assumed in the proposed radiological analysis for any DBA described in BVPS-1 UFSAR, Chapter 14, or BVPS-2 UFSAR, Chapter 15. No credit for fuel building filtration during movement of non-recently irradiated fuel is assumed in the revised FHA dose consequence analyses. The revised requirements of TS 3.9.12 (see Sections 2.3.5 and 2.4.4 of this SE) will ensure fuel building filtration during movement of recently irradiated fuel and during movement of fuel assemblies over recently irradiated fuel assemblies without reliance on the fuel building process radiation monitors. For BVPS-2 only, these monitors do initiate filtration of the fuel building air exhaust flow. The fuel building is continuously exhausted to the SLCRS filter banks. For BVPS-1 ony, during plant operation in Modes 1 through 4, SLCRS flow is automatically diverted through the main filter banks on a Containment Isolation Phase A (CIA) signal. This action ensures that SLCRS exhaust flow is filtered prior to being released to the environment. SR 4.7.8.1.c.2 demonstrates this capabiity at least once per 18 months during plant operation. For BVPS-2 only, during plant operation in Modes 1 through 4, SLCRS exhaust from plant areas contiguous to the containment is automatically diverted through the SLCRS filter train on a CIA signal to ensure that airflow from these plant areas is filtered prior to being released to the environment. SR 4.7.8.1.c.2 demonstrates this capabiity at least once per 18 months.

B. Provide indication or alarm functions relied on by operators to take manual actions that are assumed in the safety analyses for any DBA described in the BVPS-1 UFSAR, Chapter 14, or BVPS-2 UFSAR, Chapter 15. Manual actions for the fuel building ventilation are not assumed in the proposed radiological analyses.

C. Provide indication that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. These process monitors do not provide indication on the status of the reactor coolant pressure boundary. The level of radioactivity in the fuel building where the reactor pressure boundary is located, or

D. Monitor variables that have been identified as RG 1.97 Type A or Category I variable in the BVPS-1 and 2 responses to RG 1.97. The BVPS-2 RG 1.97 variable Type and Category are contained in UFSAR Table 7.5-1. The BVPS-1 RG 1.97 variable Type and Category are identified in the BVPS-1 response to GL 82-33, RG 1.97, Revision 2,

Supplemental Report, transmitted to the NRC by letter dated October 13, 1986. The fuel building process monitors are not RG 1.97 instruments.

Therefore, the NRC staff finds the proposed removal of the fuel building process radiation monitors requirements from the TSs acceptable.

5. Table 3.3-6, “Radiation Monitoring Instrumentation.” Delete the entire statement for Action 21 and replace it with the words, “This Action is not used.”

Action 21 is only referenced by the fuel building process radiation monitors that were deleted based on the above discussion. Consequently, Action 21 in not used. Therefore, the NRC staff finds acceptable the replacement of the contents of Action 21 with the words “This Action is not used.”

6. In its submittal, the licensee stated that a few editorial and format changes, changes to page numbers, grammar, capitalization, spelling, punctuation, and placement of page footers will be incorporated. The Bases Sections will also be revised to reflect the above-described changes.

The NRC staff finds the editorial and format changes do not result in any relaxation of TS requirements. Therefore, the NRC staff finds these changes acceptable.

2.2.3 Revisions to TS 3/4.6.3, “Containment Isolation Valves”

The licensee proposes to delete SR 4.6.3.1.2.c.

The requirement in SR 4.6.3.1.2.c to verify that each containment purge and isolation valve actuates to its isolation position on receipt of a Containment Purge and Exhaust isolation signal at least once per 18 months during COLD SHUTDOWN or REFUELING MODE is similar to the requirements of the proposed SR 4.9.9 (which are addressed in Section 2.2.5 of this SE). The proposed SR 4.9.9 requires, in part, that the Containment Purge and Exhaust isolation system be demonstrated operable at least once per 7 days during the movement of recently irradiated fuel and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment. The licensee believes that TS 3.9.9 is a more appropriate location for this requirement as the automatic isolation of these valves is currently only required for the mitigation of an FHA within containment. Since the proposed SR 4.9.9 essentially duplicates the requirements of SR 4.6.3.1.2.c, the deletion of SR 4.6.3.1.2.c is acceptable.

2.2.4 Revisions to TS 3/4.9.3, “Refueling Operations, Decay Time”

The word “assemblies” is added to the LCO Applicability, Action, and SR to provide a clarification that TS 3/4.9.3 applies to “irradiated fuel assemblies.” The NRC staff finds that this change is a clarification of the current TS 3/4.9.3 and does not result in any decrease in requirements. Consequently, the NRC staff finds this change acceptable.

2.2.5 Revisions to TS 3/4.9.9, “Containment Purge and Exhaust Isolation System”

1. The phrase “Core Alterations” is removed from the applicability requirements. Additionally, the phrase “and, During movement of fuel assemblies over recently

irradiated fuel assemblies within the containment" is added to the applicability requirements.

CORE ALTERATIONS is defined as the movement of fuel, sources, or reactivity control components within the reactor vessel with the reactor head removed and fuel in the vessel. The FHA is the only event during CORE ALTERATIONS that is postulated to result in fuel damage and radiological release. The fuel drop analysis determined that dropping a fuel assembly (irradiated or unirradiated) on another fuel assembly will cause fuel rod failure in the struck fuel assembly. If the struck fuel is recently irradiated fuel, a radiological release could occur. Since the phrase CORE ALTERATIONS, which includes the movement of unirradiated fuel assemblies over irradiated fuel in the vessel, is being deleted, it is necessary to modify the applicability of this specification to apply to the movement of unirradiated as well as recently irradiated fuel assemblies in the containment. The licensee's proposed revised wording for TS 3.9.9 appropriately reflects this change, and will result in the LCO being applicable during those plant evolutions that could result in an FHA. Additionally, the revised wording for containment penetration requirements of LCO 3.9.9 is consistent with TSTF-51, Revision 2, to NUREG-1431, Revision 1, which was approved by the NRC staff on October 15, 1999. Therefore, this proposed change to the applicability of LCO 3.9.9 is acceptable.

2. The proposed revised SR 4.9.9 removes the requirement to verify that the required penetrations are in their required conditions within 150 hours prior to the start of moving recently irradiated fuel assemblies or moving fuel assemblies over recently irradiated fuel assemblies within the containment. A requirement to verify that the isolation time of each Containment Purge and Exhaust Isolation System isolation valve is within limits is added.

   SR 4.0.4 requires that a surveillance be successfully performed and current prior to entering the Mode of applicability. The proposed wording of SR 4.9.9 will require that the surveillance be performed within 7 days prior to moving recently irradiated fuel or moving fuel assemblies over recently irradiated fuel assemblies within the containment. As the requirements in the modified SR 4.9.9 are essentially the same as the original, the modification is acceptable. Additionally, the addition of the requirement to verify the isolation time of each Containment Purge and Exhaust Isolation System is necessary to ensure that the requirements of the current SR 4.6.3.1.2.c are captured in the proposed SR 4.9.9.

3. The Bases of TS 3.9.9 are correspondingly revised.

2.2.6 Revisions to TS 3.9.10, "Water Level - Reactor Vessel"

1. The licensee proposes to delete the references to moving control rods in the Applicability and Action sections.

   The revised FHA analysis determined that dropping a control rod on a fuel assembly will not cause fuel cladding integrity damage. As a result, maintaining reactor vessel water level 23 feet above the fuel assemblies is not needed to mitigate an FHA when moving a control rod. Therefore, the NRC staff finds this change acceptable.

2. The licensee proposes to replace the term “pressure vessel” with “containment” in the Applicability and Action sections. This ensures that the TS is applicable whenever the potential for an FHA could occur. The NRC staff finds that this change is consistent with the revised FHA anlysis and is, therefore, acceptable.

3. The licensee proposes to add the term “irradiated” preceding “fuel assemblies” in the Applicability and Action sections.

   According to the revised FHA analysis, the dropping of a non-irradiated fuel assembly can only cause an FHA that requires the minimum water level if the non-irradiated fuel assembly strikes an irradiated fuel assembly. Therefore, adding the term “irradiated” preceding “fuel assemblies” is appropriate to make the applicability and action requirements of the TS consistent with the FHA analysis.

4. The phrase “and, During movement of fuel assemblies over irradiated fuel assemblies within the containment” is added to the Applicability requirements. The phrase “and movement of fuel assemblies over irradiated fuel assemblies within the containment” is added to the Action statement.

   The fuel drop analysis determined that dropping a fuel assembly on another fuel assembly will cause fuel rod failure in the struck fuel assembly. Maintaining reactor vessel water level 23 feet above the fuel assemblies is needed to meet the offsite and control room dose guidelines of 10 CFR 50.67. Therefore, the proposed change to the Applicability and Action statements is acceptable.

5. The licensee also proposes to modify SR 4.9.10 by removing the words “...within 2 hours prior to the start of and...” and “thereafter during movement of fuel assembly or control rods.”

   SR 4.0.4 requires that a surveillance be successfully performed and current prior to entering the Mode of applicability. The proposed wording of SR 4.9.10 will require that the surveillance be performed within 24 hours prior to moving irradiated fuel or moving fuel assemblies over irradiated fuel assemblies within the containment. As the requirements in the modified SR 4.9.10 are essentially the same as the original, the modification is acceptable. Additionally, the phrase, “thereafter during movement of fuel assembly or control rods,” is unnecessary and can be removed. The proposed wording of SR 4.9.10 will require that the SR be performed at least once per 24 hours while the TS is applicable.

6. The Bases of TS 3.9.10 are correspondingly revised.

2.2.7 Revisions to TS 3/4.9.11, “Storage Pool Level”

1. The phrase “and, During movement of fuel assemblies over irradiated fuel assemblies within the fuel storage pool” is added to the applicability requirements. The phrase “and movement of fuel assemblies over irradiated fuel assemblies within the fuel storage pool” is added to the Action statement.

The fuel drop analysis determined that dropping a fuel assembly on another fuel assembly will cause fuel rod failure in the struck fuel assembly. Maintaining reactor vessel water level 23 feet above the fuel assemblies is needed to meet the radiation exposure requirements of 10 CFR 50.67. Therefore, the proposed change to the Applicability and Action statements is acceptable.

2. The licensee proposes to delete the requirement "....and restore water level within its limits within 4 hours" in the Action statement.

   The minimum water level specified in the LCO is only required during fuel movements. If fuel movements are terminated, there is no need to restore the water level within 4 hours. The minimum water level would have to be restored prior to continuing with fuel movements. Therefore, the requirement "....and restore water level within its limits within 4 hours" is unnecessary and can be deleted. Crane operations involving heavy loads over irradiated fuel in the spent fuel pool are prohibited by administrative controls. In addition, the licensee is proposing a change to the Licensing Requirements Manual (LRM) to further restrict the movement of loads in excess of 2450 pounds over the fuel assemblies in the fuel storage pool. The current heavy load classification limit at BVPS-1 and 2, in accordance with NUREG-0612, is 2450 pounds. This change is appropriate as it clarifies the current requirement and does not result in a decrease in current requirements. Consequently, the NRC staff finds this change acceptable.

3. The licensee proposed to replace the words "...and the crane operations with load in..." in the Action statement with "within the fuel storage pool and movement of fuel assemblies over irradiated fuel assemblies within..."

   Crane operations involving heavy loads over irradiated fuel in the spent fuel pool is prohibited by administrative controls. In addition, the licensee is proposing a change to the LRM to further restrict the movement of loads in excess of 2450 pounds over the fuel assemblies in the fuel storage pool. Therefore, the deletion of "...and the crane operations with load in..." is acceptable. The addition of the phrase "within the fuel storage pool and movement of fuel assemblies over irradiated fuel assemblies within..." is necessary to ensure the assumptions of the fuel drop analysis are met.

4. The Bases of TS 3.9.11 are correspondingly revised.

2.3 Revisions to the BVPS-1 TS Only

2.3.1 Revisions to TS 3/4.3, "Instrumentation" include:

1. Table 3.3-6, "Radiation Monitoring Instrumentation," Item 1.b.i, "Area Monitors, Containment Purge & Exhaust Isolation (RMVS 104 A & B)." Change minimum channels operable from 1 to 2.

The current BVPS-1 TSs require that, while the plant is in Mode 6, a minimum of one channel of area radiation monitoring instrumentation in the containment purge and exhaust isolation system be operable. The proposed amendment changes the number of minimum operable channels from one to two, and changes the applicable mode from Mode 6 to a period during which recently irradiated fuel assemblies are being moved within the containment and/or during

which fuel assemblies are being moved over recently irradiated fuel assemblies. In its submittal, the licensee stated that changing minimum channels operable from one to two ensures that, during an FHA inside the containment involving recently irradiated fuel assemblies, redundant area radiation monitoring channels are available to detect elevated levels of airborne radiation in the containment, initiate isolation of the containment purge, and exhaust the containment penetrations. The proposed change is also consistent with the NUREG-1431, “Standard Technical Specifications - Westinghouse Plants,” Revision 1, Specification 3.3.6, “Containment Purge and Exhaust Isolation Instrumentation.” Changing the minimum number of operable channels from one to two is conservative and, therefore, is acceptable to the NRC staff.

2. Table 3.3-6, change “6” to table notation (2) in Applicable Modes column.

3. Table 3.3-6, revise current notation (2) to read: “During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.”

4. Table 4.3-3, “Radiation Monitoring Instrumentation Surveillance Requirements,” Item 1.b.i, “Area Monitors, Containment Purge & Exhaust Isolation (RMVS 104 A & B).” Change mode in which surveillance is required from “Mode 6” to a footnote designated by “**.”

5. Table 4.3-3, revise the current footnote designated “**” to read: “During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.”

6. Table 3.3-6, revise the current notation 4 to read: “During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.”

7. Table 3.3-6, “Radiation Monitoring Instrumentation,” revise Action 41 by adding the words “assemblies and movement of fuel assemblies over irradiated fuel assemblies,” after the word “fuel” in a).2 and in b).2.

8. Table 4.3-3, revise the current footnote ## to read: “During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.”

In its submittal, the licensee stated that during Modes 5 and 6, while fuel handling is not being performed, these monitors will not be required to be operable. This is because, during these periods, the potential for radioactive releases is minimized and operator action is sufficient to ensure that the post-accident offsite doses are maintained within the regulatory limits. The addition of the term “During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies” is necessary based on the results of the fuel drop analysis as discussed in Section 2.1 of this SE. The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod rupture. Additionally, the movement of a non-irradiated fuel assembly could result in a radiological release that might require the use of the control room habitability systems if this assembly were to strike an irradiated fuel assembly. The proposed Modes will continue to require these monitors to be operable when there is a potential for an FHA, that is, “during movement of recently irradiated fuel assemblies (within the containment) and during the movement of fuel assemblies over

recently irradiated fuel assemblies (within the containment)." Therefore, the proposed changes to the LCO Applicabilities and Actions, including the referenced footnotes are acceptable to the NRC staff. Since surveillance requirements are not needed for any equipment that is not required to be operable, the proposed change in the Mode in which the surveillance is required is consistent with the operability requirement for these monitors and, therefore, is acceptable to the NRC staff. Therefore, the changes discussed above are acceptable.

9. Table 3.3-6, "Radiation Monitoring Instrumentation," Item 1.c, "Area Monitors, Control Room Isolation (RM-RM-218 A & B)." In the Applicable Modes column, delete the words, "in either unit."

10. Table 4.3-3, "Radiation Monitoring Instrumentation Surveillance Requirements," Item 1.c, "Area Monitors, Control Room Isolation (RM-RM-218 A & B)." Delete the words "(in either unit)," in the Modes in which Surveillance Required column.

BVPS-1 and 2 control rooms are open to each other and share the same pressure envelope but each unit has its own control room emergency habitability systems. Both BVPS-1 and 2 have TSs for their respective control room habitability systems which are sufficient to protect the operators at both units during evolutions that may result in an FHA. Therefore, the stipulation "in either unit" is not necessary and the NRC staff finds the deletion of this phrase acceptable.

2.3.2 Revisions to TS 3/4.7.7, "Control Room Emergency Habitability Systems"

The licensee proposes to make the following changes to the applicability and required actions of LCO 3.7.7.1:

1. In the applicability statement:

    A. Delete the requirement that this TS is applicable if either unit meets the Mode applicability criteria.

    BVPS-1 and 2 control rooms are open to each other and share the same pressure envelope but each unit has its own control room habitability systems. Both BVPS-1 and 2 have TSs for their respective control room habitability systems which are sufficient to protect the operators at both units during evolutions which may result in an FHA. Therefore, the stipulation "with either unit" is not necessary and we find the deletion to be acceptable.

    B. Delete the requirement of "movement of loads over irradiated fuel at either unit." Add the requirement that the TS is applicable during the movement of fuel assemblies over irradiated fuel assemblies.

    The movement of heavy loads over the reactor core with the potential to damage fuel assemblies has been evaluated by the licensee and safe load paths have been incorporated within plant procedures as required by NUREG-0612. Crane operations involving heavy loads over irradiated fuel in the spent fuel pool are prohibited by administrative controls as described in applicable guidance, NUREG-0612 and GL 81-07, "Control of Heavy Loads." In addition, the licensee is proposing a change to the LRM to further restrict the movement of loads in excess of 2450 pounds over the

fuel assemblies in the fuel storage pool. Analysis shows that dropping a load with less impact energy than a fuel assembly will not cause significant deformation of the fuel storage racks which protect the spent fuel. For the analysis, the licensee used a total fuel assembly weight of 2500 pounds, which is 50 pounds over the heavy load classification of 2450 pounds. The 50 pounds was added for conservatism and extra margin in case the total fuel assembly weight increased due to changes in fuel assembly design. Based on the above actions taken by the licensee, the deletion of the requirement of “movement of loads over irradiated fuel at either unit” is acceptable. The revised fuel drop analysis determined that dropping a fuel assembly on another fuel assembly will cause fuel rod failure in the struck fuel assembly. Adding the requirement that the TS is applicable during the movement of fuel assemblies over irradiated fuel assemblies is necessary to meet the fuel drop analysis.

C. Delete the reference to TS 3.9.15.

This change updates TS 3.7.7.1 to reflect the deletion of TS 3.9.15. This is an administrative change, since the requirements of TS 3.3.15 are now included in TS 3.7.7.1 (see discussion of deletion of TS 3.9.15 in Section 2.3.7 in this SE). The NRC staff finds this change acceptable.

2. Action statements b.1 and c.1 are changed to reflect the new applicability requirements as discussed above. These changes reflect the revisions to the applicability statement and are acceptable.

3. LCO 3.7.7.1.c is revised by replacing the words “or closed” with “or OPERABLE by being secured in a closed position with power removed” for the required condition of the series normal air intake and exhaust isolation dampers for both units.

   The revision to LCO 3.7.7.1.c specifies the required condition of the series normal air intake and exhaust isolation dampers for both units. The revision states that if the dampers are closed with power removed, the system is still operable. No relaxation of requirements results from this revision; therefore, the NRC staff finds this change acceptable.

4. Footnote “*” is revised to specify that emergency power for only one train of dampers and fans in a pressurization filtration unit is “required in Modes 5, 6 and with no fuel in the reactor pressure vessel.” The revision to this footnote is a restating of the requirement to clearly identify what is required in Modes 5 and 6. The phrase “with no fuel in the reactor pressure vessel” is necessary since that condition takes the plant out of Modes 5 and 6 but fuel handling activities in the fuel building to which TS 3.7.7.1 is applicable may still be occurring. Consequently, the NRC staff finds this change acceptable.

5. Footnote “***,” stating that the automatic isolation of the series normal intake and exhaust isolation dampers on a Containment Isolation B signal is only required in Modes 1 through 4 is added to LCO 3.7.7.1.c. The addition of this footnote is acceptable as it clarifies the original requirement.

6. The Bases of TS 3.7.7.1 are revised to reflect the changes discussed above.

2.3.3 Revisions to TS 3/4.8, “Electrical Power Systems”

1. For LCO 3.8.1.2, “A. C. Sources - Shutdown,” LCO 3.8.2.2, “A. C. [alternating current] Distribution - Shutdown,” and LCO 3.8.2.4, “D. C. [direct current] Distribution - Shutdown,” the LCO Applicability pertaining to fuel movement will be revised to state the following: “During movement of irradiated fuel assemblies and During movement of fuel assemblies over irradiated fuel assemblies.” The LCO Action requirements will also be revised by adding the word “assemblies” following the word “fuel” and replacing the word “loads” with the word “fuel assemblies.”

The licensee stated that the proposed revisions of the BVPS-1 LCO applicability and action requirements, specified in LCOs 3.8.1.2, 3.8.2.2, and 3.8.2.4 for the electrical power requirements will ensure that the requirements of these specifications apply and appropriate actions are taken during fuel handling activities that involve the potential need for ac and dc power systems to mitigate the consequences of an FHA in order to meet the radiation exposure limits specified in 10 CFR 50.67. The addition of “and During movement of fuel assemblies over irradiated fuel assemblies” is necessary based on the results of the fuel drop analysis. The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod failure. The movement of a non-irradiated fuel assembly could result in a radiological release that may require the use of the control room habitability systems if this assembly were to strike an irradiated fuel assembly. Therefore, the adoption of the ISTS LCO applicability wording of “During movement of irradiated fuel assemblies” is not sufficient to meet the BVPS-1 and 2 fuel drop analysis.

The control room habitability system (i.e., the pressurization bottles, emergency supply fans and damper isolation) is not required to mitigate the consequences of an FHA (based on the proposed radiological analysis) in order to maintain the radiation doses to personnel occupying the control room to within the limits of 10 CFR 50.67. The revised BVPS-1 analysis assumes only that a 30-minute purge of the control room atmosphere occurs following radioactivity release termination. In order to ensure that sufficient electrical power is available to conduct a purge of the control room following an FHA and as a conservative measure to support the control room habitability systems, the term “recently” will not be added to the BVPS-1 LCO applicability and action requirements for electrical power specified in LCOs 3.8.1.2, 3.8.2.2, and 3.8.2.4. The basis for not adding the term “recently” to these requirements is also a conservative measure. The requirement to maintain the control room habitability systems will provide additional assurance that a sufficient level of control room ventilation equipment is available to conduct the 30-minute purge of the control room following an FHA.

The proposed deletion of the Mode applicability of “during movement of loads over irradiated fuel assemblies” is consistent with the ISTS. For the fuel storage building, the fuel storage racks are designed to withstand the dropping of a fuel assembly and remain capable of performing their safety function to prevent a criticality event. Separation between the dropped fuel assembly and the top of the active fuel height of the stored fuel assembly is adequate to preclude interaction. Thus, loads that have an impact energy that is less than or equal to the impact energy of a dropped fuel assembly will not result in significant deformation of the fuel storage racks.

Crane operations involving heavy loads (greater than 2450 pounds) over irradiated fuel stored in the spent fuel pool are prevented by administrative controls in compliance with NUREG-0612. These administrative controls are implemented in accordance with the guidance of NUREG-0612 and associated GL 81-07. The BVPS-1 and 2 administrative controls for moving heavy loads provide adequate assurance that operation of the fuel building crane will not include lifting heavy loads that have the potential to damage stored fuel assemblies.

The movement of loads over irradiated fuel within the containment is addressed by the development of safe load paths consistent with the guidance provided in NUREG-0612 and the current requirement to require FHA mitigation systems when performing core alterations. These requirements provide additional assurance that FHA mitigation systems are operable when moving loads (fuel assemblies) within the reactor vessel and the potential exists for an FHA. The proposed changes to the applicable LCOs will continue to require that FHA mitigation systems be operable during movement of loads (fuel assemblies) over the reactor vessel containing recently irradiated fuel assemblies. The movement of heavy loads over the reactor core with the potential to damage fuel assemblies in the reactor core has been evaluated and safe load paths have been incorporated into the BVPS-1 and 2 plant procedures. Changes to established safe load paths can only be made in accordance with the provisions of 10 CFR 50.59.

In addition, the proposed change to the LRM, Section 7.1, “Crane Travel - Spent Fuel Storage Pool Building,” will further restrict the movement of loads in excess of 2450 pounds over fuel assemblies in the fuel storage pool. A weight of greater than 2450 pounds is the current heavy load classification limit at BVPS-1 and 2 in accordance with NUREG-0612.

The NRC staff finds that the changes are consistent with the revised FHA dose consequence analyses and are conservative. The changes are also consistent with the ISTS. Consequently, the NRC staff finds the changes acceptable.

2. Changes to the Bases for TS Sections 3.8.1.2, 3.8.2.2, and 3.8.2.4 are made to reflect the changes discussed above. These changes are acceptable.

2.3.4 Revisions to TS 3/4.9.4, “Containment Penetrations”

1. LCO 3.9.4.c.2 is being revised to state:

   “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 ≤7500 cfm [cubic feet per minute] to at least one OPERABLE filtered Supplemental Leak Collection and Release System (SLCRS) train.”

   The equivalent BVPS-2 LCO was previously approved by License Amendment No. 116, dated September 28, 2000 (ADAMS Accession No. ML003749558). This change is consistent with the revised FHA analysis. Consequently, the NRC staff finds this change acceptable.

2. The term CORE ALTERATIONS is removed from the Applicability statement. Additionally, the applicability and action statements are being revised by the addition of

"...movement of fuel assemblies over recently irradiated fuel assemblies within the containment."

CORE ALTERATIONS is defined as the movement of fuel, sources, or reactivity control components within the reactor vessel with the reactor head removed and fuel in the vessel. The FHA is the only event during CORE ALTERATIONS that is postulated to result in fuel damage and radiological release. The fuel drop analysis determined that dropping a fuel assembly onto another fuel assembly will cause fuel rod failure in the struck fuel assembly. If the struck fuel is recently irradiated fuel, a radiological release in excess of 10 CFR 50.67 dose limits could occur. Since CORE ALTERATIONS, which includes the movement of unirradiated fuel assemblies over irradiated fuel in the vessel, is being deleted, it is necessary to modify the applicability of this specification to apply to the movement of unirradiated as well as recently irradiated fuel assemblies in the containment. The licensee's proposed revised wording for TS 3.9.4 appropriately reflects this change, and will result in the LCO being applicable during those plant evolutions that could result in an FHA. Additionally, the revised wording for containment penetration requirements of LCO 3.9.4 is consistent with TSTF-51, Revision 2, to NUREG-1431, Revision 1, which was approved by the NRC staff on October 15, 1999. Therefore, this proposed change to the applicability of LCO 3.9.4 is acceptable.

3. SR 4.9.4.1 will be modified by removing the requirement to verify that the required penetrations are in their required conditions within 150 hours prior to the start of moving recently irradiated fuel assemblies or moving fuel assemblies over recently irradiated fuel assemblies within the containment.

   SR 4.0.4 requires that a surveillance be successfully performed and current prior to entering the Mode of applicability. The proposed wording of SR 4.9.4.1 will require that the surveillance be performed within 7 days prior to moving recently irradiated fuel or moving fuel assemblies over recently irradiated fuel assemblies within the containment. As the requirements in the modified 4.9.4.1 are essentially the same as the original, the modification is acceptable.

4. The proposed revision to SR 4.9.4.2.c will require that the required portions of the SLCRS will be demonstrated OPERABLE per Specification 4.7.8.1 with the exception to SR 4.7.8.1.c.2. SR 4.7.8.1.c.2 is only applicable in Modes 1-4. The revision is acceptable as it clarifies the original requirement.

5. The Bases of TS 3.9.4 are correspondingly revised.

2.3.5 Revisions to TS 3/4.9.12, " Fuel Building Ventilation System - Fuel Movement."

The licensee proposes to modify TS 3.9.12 by:

1. Changing the LCO and Applicability. The licensee proposes to change the words from:

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

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

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

Applicability: When irradiated fuel which has decayed less than 60 days is in the fuel storage pool."

to:

"The fuel building portion of the Supplemental Leak Collection and Release System (SLCRS) shall be OPERABLE and operating with fuel building exhaust flow discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers.

Applicability: During the movement of recently irradiated fuel assemblies within the fuel storage pool and during movement of fuel assemblies over recently irradiated fuel assemblies within the fuel storage pool"

According to the revised FHA analysis, the only FHA in the fuel building that could exceed 10 CFR 50.67 dose limitations can only occur during the movement of recently irradiated fuel assemblies or fuel assemblies over recently irradiated fuel assemblies within the fuel storage pool. The revised LCO and Applicability requirements reflect the FHA analysis by requiring at least one train of the fuel building portion of the SLCRS to be operable and operating with fuel building exhaust flow discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers when those conditions exist. The LCO ensures that no failures preventing operation of the SLCRS will go undetected when the potential for a design-basis FHA exists. This change also clarifies that the fuel building ventilation system is part of SLCRS.

The change in applicability is consistent with the revised FHA analysis. According to the revised FHA analysis, an FHA with fuel decayed more than 100 hours will not exceed the 10 CFR 50.67 dose limits. Therefore, the NRC staff finds this change acceptable.

See Section 2.3.2 of this SE for a discussion of the deletion of crane operations involving loads over the spent fuel pool.

2. Changing the Action statement to reflect the change in LCO applicability. The NRC staff finds that the changes to the Action statement are consistent with the revised FHA analysis and are, therefore, acceptable.

3. Renumbering SR 4.9.12 to SR 4.9.12.1 and revising the wording of the surveillance to be consistent with the LCO. These changes are administrative and reflect accurately the other changes made to this TS section and are, therefore, acceptable.

4. Modifying SR 4.9.12.1 by removing the words "...within 2 hours prior to the initiation of and..."

   SR 4.0.4 requires that a surveillance be successfully performed and current prior to entering the Mode of applicability. The proposed wording of SR 4.9.12.1 will require that the surveillance be performed within 12 hours prior to moving recently irradiated fuel or

moving fuel assemblies over recently irradiated fuel assemblies within the fuel storage pool. The intent of the original requirement is maintained; therefore, this change is acceptable.

5. Adding SR 4.9.12.2, which is moved from SR 4.9.13. SR 4.9.13 is retained as discussed in Section 2.3.6 of this SE. TS 3/4.9.12.2 is the appropriate location of this SR. Consequently this change is acceptable.

6. Adding a footnote to SR 4.9.12 which states “The fuel building doors may be open for entry and exit.”

   This footnote is necessary to allow the plant area doors to be opened for entry and exit. Also, the fuel building portion of SLCRS maintains the fuel building at a slight negative pressure, which would limit any release through the doors if opened. The addition of this footnote is necessary to allow access to the fuel building while this TS is applicable.

7. Revising the Bases of TS 3.9.12. The NRC staff finds that these changes appropriately reflect the changes discussed above and are acceptable.

2.3.6 Revisions to TS 3/4.9.13, “Fuel Building Ventilation System - Fuel Storage”

1. The licensee proposes to delete SR 4.9.13.a and b.1 and move the remaining SR 4.9.13 to TS 3.9.12 as SR 4.9.12.2.

   A. SR 4.9.13.a requires initiation of flow through the fuel building ventilation system at least once every 31 days and verification that the system operates at least 15 minutes.

      This requirement is contained in the current SR 4.9.13.c and is being retained as SR 4.9.12.2.b.

   B. SR 4.9.13.b.1 requires verification every 18 months that the fuel building ventilation system redirects its exhaust through the SLCRS filter banks and charcoal adsorbers on a high-high radiation signal.

      The requirement in this SR is unnecessary as TS 3.9.12 requires the fuel building portion of SLCRS to be in this line-up prior to initiating activities which could result in a design-basis FHA. The LCO applicability, “Whenever irradiated fuel is in the storage pool,” is not relevant to the mitigation of a DBA. According to the revised FHA analysis, an FHA in the fuel building can only occur during the movement of fuel assemblies which is covered by TS 3.9.12. Therefore, the NRC staff finds the deletion of this requirement acceptable.

2. The licensee also proposes to delete the remaining portions of TS 3.9.13.

   TS 3.9.13 is no longer needed after the transfer of SR 4.9.13 to TS 3.9.12. The LCO applicability “Whenever irradiated fuel is in the storage pool” is not relevant to the mitigation of a DBA. According to the revised FHA analysis, an FHA in the fuel building can only occur during the movement of fuel assemblies. FHA mitigation systems are

only required during fuel movement. The movement of fuel assemblies in the fuel building is addressed in TS 3.9.12. Therefore, the NRC staff finds the deletion of TS 3.9.13 acceptable.

3. The proposed revision to the current SR 4.9.13.c will be reflected in the new SR 4.9.12.2. The proposed revision will result in the requirement that the fuel building portions of the SLCRS be demonstrated OPERABLE per Specification 4.7.8.1 with the exception to item 4.7.8.1.c.2 (SR 4.7.8.1.c.2 is only applicable in Modes 1-4). This change clarifies the original requirement and the NRC staff, therefore, finds the change acceptable.

4. The Bases of TS 3.9.13 are correspondingly deleted.

2.3.7 Revisions to TS 3/4.9.15, “Control Room Emergency Habitability Systems”

1. The licensee proposes to delete this specification.

BVPS-1 and 2 control rooms are open to each other and share the same pressure envelope but each unit has its own control room emergency habitability systems. TS 3.9.15 sets requirements for the BVPS-1 Control Room Emergency Habitability Systems when both units are in Modes 5 or 6. The licensee considers the requirements of TS 3.9.15 to be no longer necessary since the operability of the control room emergency habitability systems will not be required in Modes 5 or 6 consistent with the assumptions of the revised FHA dose consequence analysis. Both BVPS-1 and 2 have other TS requirements addressing their respective control room emergency habitability systems which are applicable during fuel handling evolutions which could potentially result in an FHA. The licensee considers the individual unit’s TSs for control room emergency habitability systems sufficient to protect the operators at either unit. TS 3.7.7.1 specifies the requirement for the BVPS-1 control room emergency habitability system to meet the assumptions for control room emergency habitability systems credited in the revised FHA dose consequence analysis. The NRC staff finds that these changes are consistent with the revised FHA dose consequence analyses. Additionally, existing TSs provide sufficient requirements for control room emergency habitability systems. Consequently, the NRC staff finds the changes acceptable.

2. The Bases of TS 3.9.15 are correspondingly deleted.

2.4 Revisions to the BVPS-2 TSs Only

2.4.1 Revisions to TS 3/4.3, “Instrumentation”

1. Table 3.3-6, “Radiation Monitoring Instrumentation.” Revise current notation (4) to read, “During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.”

2. Table 3.3-6, revise current notation (5) to read, “During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.”

3. Table 4.3-3, revise the current notation designated by "##" to read: "During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies."

4. Table 4.3-3, revise current notation designated by "###" to read: "During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment."

In its submittal, the licensee stated that during Modes 5 and 6, while fuel handling is not being performed, these monitors will not be required to be operable. This is because, during these periods, the potential for radioactive releases is minimized and operator action is sufficient to ensure that the post-accident offsite doses are maintained within the regulatory limits. The addition of the term "During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies" is necessary based on the results of the fuel drop analysis as discussed in Section 2.1 of this SE. The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod rupture. Additionally, the movement of a non-irradiated fuel assembly could result in a radiological release that might require the use of the control room habitability systems if this assembly were to strike an irradiated fuel assembly. The proposed Modes will continue to require these monitors to be operable when there is a potential for an FHA, that is, "during movement of recently irradiated fuel assemblies (within the containment) and during the movement of fuel assemblies over recently irradiated fuel assemblies (within the containment)." Therefore, the proposed changes to the Table notations, including the referenced footnotes are acceptable to the NRC staff. Since surveillance requirements are not needed for any equipment that is not required to be operable, the proposed change in the mode in which the surveillance is required is consistent with the operability requirement for these monitors and, therefore, is acceptable to the NRC staff. Therefore, the changes discussed above are acceptable. These changes include the stated revisions to Table 3.3-6 notations (4) and (5) and Table 4.3-3 notations "##" and "###."

5. Replace the entire statement of current Table 3.3-6 notation (2) with the words, "Not used."

Table 3.3-6 notation (2) is currently referenced in the Applicable Modes column and specified that certain instruments need to be operable "with irradiated fuel in the storage pool." Consistent with the revised FHA dose consequence analysis, a postulated FHA can occur only when a fuel assembly is being moved within the fuel building not merely when irradiated fuel is in the storage pool, i.e., operability is necessary when an FHA could occur. Table notation (2) is not relevant to the mitigation of an FHA. Therefore, the NRC staff finds the replacement of the contents of Table notation (2) with the word "Not used" acceptable.

6. Table 4.3-3, "Radiation Monitoring Instrumentation Surveillance Requirements," delete the Footnote designated as, " ** ."

Table footnote "**" was only referenced by Unit 2, Item 2.b.i, "Process Monitors, Fuel Building Vent Gaseous Activity (Xe -133) (2RMF - RQ301B)," and Item 2.b.ii, "Particulate (I - 131) (2RMF - RQ301A)." The NRC staff's evaluation approving the deletion of TS requirements for these radiation process monitors is documented in Section 2.2.2 of this SE. Therefore, the deletion of Footnote "**" is acceptable to the NRC staff.

2.4.2 Revisions to TS 3/4.7.7, “Control Room Emergency Air Cleanup and Pressurization System”

1. The licensee proposes to change the applicability of this specification from “All modes” to “Modes 1, 2, 3, 4, and During movement of recently irradiated fuel assemblies, and During movement of fuel assemblies over recently irradiated fuel assemblies.” Additionally, “Modes 5 and 6" in the revised Action section is replaced with “During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.”

   The only DBA postulated to occur when the plant is in Modes 5 or 6 is an FHA. The only postulated FHA to cause radiation releases that exceed 10 CFR 50.67 limits occurs during movement of recently irradiated fuel assemblies or during movement of fuel assemblies over recently irradiated fuel assemblies. Therefore, the Control Room Emergency Air Cleanup and Pressurization System is not needed for FHA mitigation in Modes 5 or 6 unless recently irradiated fuel is being moved or fuel assemblies are being moved over recently irradiated fuel assemblies. Therefore, the NRC staff finds this change in the applicability of TS 3.7.7 acceptable.

2. The phrase “...CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel and movement of loads over irradiated fuel” in the required actions is replaced with “movement of recently irradiated fuel assemblies and movement of fuel assemblies over recently irradiated fuel assemblies.”

   CORE ALTERATIONS is defined as the movement of fuel, sources, or reactivity control components within the reactor vessel with the reactor head removed and fuel in the vessel. Of the possible events resulting from CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel and movement of loads over irradiated fuel, only the FHA is postulated to result in fuel damage and radiological release. The fuel drop analysis determined that dropping a fuel assembly on another fuel assembly will cause fuel rod failure in the struck fuel assembly. If the struck fuel is recently irradiated fuel, a radiological release in excess of 10 CFR 50.67 dose limits could occur. Since the phrase CORE ALTERATIONS, which includes the movement of unirradiated fuel assemblies over irradiated fuel in the vessel, is being deleted, it is necessary to modify the applicability of this specification to apply to the movement of unirradiated as well as recently irradiated fuel assemblies in the containment. The licensee’s proposed revised wording for TS 3.9.4 appropriately reflects this change, and will result in the LCO being applicable during those plant evolutions that could result in an FHA. Additionally, the revised wording for containment penetration requirements of LCO 3.7.7 is consistent with TSTF-51, Revision 2, to NUREG-1431, Revision 1, which was approved by the NRC staff on October 15, 1999. Therefore, the NRC staff finds this change acceptable.

3. A new footnote, Footnote “***”, is added to LCO 3.7.7.1.c to clarify that control room isolation on a Containment Isolation B signal is only required in Modes 1 through 4 for operability. The NRC staff finds that this change clarifies the TS requirement and is acceptable.

4. A revised Footnote “**” specifies that emergency power for only one train of dampers and fans in a pressurization filtration unit is required in “Mode 5, 6 and with no fuel in the reactor pressure vessel.” The revision to this footnote is a restating of the requirement to clearly identify what is required in Modes 5 and 6. The phrase “With no fuel in the reactor pressure vessel” is necessary since that condition takes the plant out of Modes 5 and 6, but fuel handling activities in the fuel building applicable to TS 3.7.7 may still be occurring. The NRC staff finds that this change clarifies the TS requirement and is acceptable.

5. The Bases of TS 3.7.7 are correspondingly revised.

2.4.3 TS 3/4.8, “Electrical Power Systems”

1. LCO 3.8.1.2, “A.C. Sources - Shutdown,” LCO 3.8.2.2, “A.C. Distribution - Shutdown,” and LCO 3.8.2.4, “D.C. Distribution - Shutdown,” the LCO applicability pertaining to fuel movement will be revised to state the following: “During movement of recently irradiated fuel assemblies and During movement of fuel assemblies over recently irradiated fuel assemblies.” The LCO Action requirements will also be revised by adding the word “assemblies” following the word “fuel,” adding the word “recently” preceding the word “irradiated,” and replacing the word “loads” with the word “fuel assemblies.” For LCO 3.8.2.2 only, the word “than” will be added to the LCO Action requirements following the word “less.”

The licensee stated that the proposed revisions of the BVPS-2 LCO applicability and action requirements, specified in LCOs 3.8.1.2, 3.8.2.2, and 3.8.2.4 for the electrical power requirements, will ensure that the requirements of these specifications apply and appropriate actions are taken during fuel handling activities that involve the potential need for ac and dc power systems to mitigate the consequences of an FHA in order to meet the radiation exposure limits specified in 10 CFR 50.67. The addition of “and During movement of fuel assemblies over recently irradiated fuel assemblies” is necessary based on the results of the fuel drop analysis. The fuel drop analysis has determined that damage to the struck fuel assembly will result in fuel rod failure. The movement of a non-irradiated fuel or non-recently irradiated fuel assembly could result in a radiological release that may require the use of the control room habitability systems if this assembly were to strike a recently irradiated fuel assembly. Therefore, the adoption of the ISTS LCO applicability wording of “During movement of irradiated fuel assemblies” is not sufficient to meet the BVPS-1 and 2 fuel drop analysis. Therefore, the licensee proposed to add the term “and During movement of fuel assemblies over recently irradiated fuel assemblies.” The NRC staff finds these revisions conservative and consistent with the revised FHA dose consequence analysis and therefore, acceptable.

The addition of the term “recently” to the above LCO applicability is based on the radiological analysis and guidance provided in TSTF-51, Revision 2. The purpose of the addition of the term “recently” is to establish a point where operability of those systems typically used to mitigate the consequences of an FHA is no longer required to meet the radiation exposure limits specified in 10 CFR 50.67. The ac and dc power systems are no longer required to be operable to mitigate an FHA once the irradiated fuel has undergone 100 hours of radioactive decay.

The proposed deletion of the Mode applicability of “during movement of loads over irradiated fuel assemblies” is consistent with the ISTS. For the fuel storage building, the fuel storage racks are designed to withstand the dropping of a fuel assembly and remain capable of performing their safety function to prevent a criticality event. Separation between the dropped fuel assembly and the top of the active fuel height of the stored fuel assembly is adequate to preclude interaction. Thus, loads that have an impact energy that is less than or equal to the impact energy of a dropped fuel assembly will not result in significant deformation of the fuel storage racks.

Crane operations involving heavy loads (greater than 2450 pounds) over irradiated fuel stored in the spent fuel pool are prevented by administrative controls in compliance with NUREG-0612. The administrative controls are required to be implemented by NUREG-0612 and associated GL 81-07. The BVPS-1 and 2 administrative controls for moving heavy loads provide adequate assurance that operation of the fuel building crane will not include lifting heavy loads that have the potential to damage stored fuel assemblies.

The movement of loads over irradiated fuel within the containment is addressed by the development of safe load paths consistent with the guidance provided in NUREG-0612 and the current requirement to require FHA mitigation systems when moving fuel assemblies within containment. These requirements provide additional assurance that FHA mitigation systems are operable when moving loads within the reactor vessel and the potential exists for a fuel handling accident. The proposed changes to the applicable LCOs will continue to require that FHA mitigation systems be operable during movement of fuel assemblies over the reactor vessel containing recently irradiated fuel assemblies. The movement of heavy loads over the reactor core with the potential to damage fuel assemblies in the reactor core has been evaluated and safe load paths have been incorporated into the BVPS plant procedures. Changes to established safe load paths can only be made in accordance with the provisions of 10 CFR 50.59.

In addition, the proposed change to LRM, Section 7.1, “Crane Travel - Spent Fuel Storage Pool Building,” will further restrict the movement of loads in excess of 2450 pounds over fuel assemblies in the fuel storage pool. A weight of greater than 2450 pounds is the current heavy load classification limit at BVPS-1 and 2 in accordance with NUREG-0612.

On the basis of its review, the NRC staff finds the changes to TS Sections 3.8.1.2, 3.8.2.2, and 3.8.2.4 acceptable because they are conservative and consistent with the revised FHA analysis.

2. TS Bases 3/4.8.1, 3/4.8.2 are revised to reflect the changes to TS Sections 3.8.1.2, 3.8.2.2, and 3.8.2.4 as discussed above.

2.4.4 TS 3/4.9.12, “ Fuel Building Ventilation System - Fuel Movement”

The licensee proposes to modify TS 3.9.12 by:

1. Changing the LCO and Applicability. The licensee proposes to change the words from:

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

a. Fuel movement within the spent fuel pool, or

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

Applicability: When irradiated fuel which has decayed less than 60 days is in the fuel storage pool."

to:

"The fuel building portion of the Supplemental Leak Collection and Release System (SLCRS) shall be OPERABLE and operating with fuel building exhaust flow discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers.

Applicability: During the movement of recently irradiated fuel assemblies within the fuel storage pool and during movement of fuel assemblies over recently irradiated fuel assemblies within the fuel storage pool"

According to the revised FHA analysis, the only FHA in the fuel building that potentially could exceed 10 CFR 50.67 dose limitations can only occur during the movement of recently irradiated fuel assemblies or fuel assemblies over recently irradiated fuel assemblies within the fuel storage pool. The revised LCO and Applicability requirements reflect the FHA analysis by requiring at least one train of the fuel building portion of the SLCRS to be operable and operating with fuel building exhaust flow discharging through at least one train of the SLCRS HEPA filters and charcoal adsorbers when those conditions exist. The LCO ensures that no failures preventing operation of the SLCRS will go undetected when the potential for a design-basis FHA exists. This change also clarifies that the fuel building ventilation system is part of SLCRS.

The change in applicability is consistent with the revised FHA analysis. According to the revised FHA analysis, an FHA with fuel decayed more than 100 hours will not exceed the 10 CFR 50.67 dose limits. Therefore, this change is acceptable.

See Section 2.3.2 of this SE for a discussion of the deletion of crane operations involving loads over the spent fuel pool.

2. Changing the Action statement to reflect the change in LCO applicability. The NRC staff finds that these changes accurately reflect the revised FHA analysis and are acceptable.

3. Renumbering SR 4.9.12 to SR 4.9.12.1 and revising the wording of the surveillance to be consistent with the LCO. This is an administrative change and is acceptable.

4. Modifying the proposed SR 4.9.12.1 by removing the words "...within 2 hours prior to the initiation of and..."

SR 4.0.4 requires that a surveillance be successfully performed and current prior to entering the Mode of applicability. The proposed wording of SR 4.9.12.1 will require that the surveillance be performed within 12 hours prior to moving recently irradiated fuel or moving fuel assemblies

over recently irradiated fuel assemblies within the fuel storage pool. The intent of the original requirement is maintained; therefore, this change is acceptable.

5. Adding a footnote to SR 4.9.12, which states "The fuel building doors may be open for entry and exit."

This footnote is necessary to allow the plant area door to be opened for entry and exit. If the doors are not held open, the air infiltration into the fuel building is negligible. The fuel building portion of SLCRS maintains the fuel building at a slight negative pressure.

6. Revising the Bases of TS 3.9.12. The NRC staff finds that these changes appropriately reflect the changes discussed above and are acceptable.

2.4.5 TS 3/4.9.13, " Fuel Building Ventilation System - Fuel System."

1. The proposed revision to the current SR 4.9.13 will require that the required portions of the SLCRS will be demonstrated OPERABLE per Specification 4.7.8.1 with the exception to item 4.7.8.1.c.2. SR 4.7.8.1.c.2 is only applicable in Modes 1-4. The NRC staff finds this revision acceptable because it clarifies the original requirement.

2. The licensee proposes to move SR 4.9.13 to TS 3.9.12 as SR 4.9.12.2 and delete TS 3.9.13.

   TS 3.9.13 is no longer needed after the transfer of SR 4.9.13 to TS 3.9.12. The LCO applicability "Whenever irradiated fuel is in the storage pool." is not relevant to the mitigation of a DBA. According to the revised FHA analysis, an FHA can only occur during the movement of fuel assemblies in the fuel building. FHA mitigation systems are only required during fuel movement. The movement of fuel assemblies in the fuel building is addressed in TS 3.9.12. Therefore, the NRC staff finds the deletion of TS 3.9.13 acceptable.

3. The Bases of TS 3.9.13 are correspondingly deleted.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the *Federal Register* on August 23, 2001 (66 FR 44385). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: August 30, 2001

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

FIRSTENERGY NUCLEAR OPERATING COMPANY, ET AL.

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 241 and 121 to Facility Operating License Nos. DPR-66 and NPF-73, respectively, issued to FirstEnergy Nuclear Operating Company, et al. (the licensee), which revised the Technical Specifications (TSs) and authorized revisions to the Updated Final Safety Analysis Report (UFSAR) for operation of Beaver Valley Power Station, Unit Nos. 1 and 2, located in Shippingport, Pennsylvania. The amendment is effective as of the date of issuance.

The amendment authorized revisions to the BVPS-1 and 2 UFSAR design-basis fuel handling accident (FHA) dose consequence analyses. The amendment also revised the BVPS-1 and 2 TSs associated with the requirements for handling irradiated fuel assemblies in the reactor containment and fuel building and the TS requirements associated with ensuring that UFSAR safety analysis assumptions are met for a postulated FHA. The term "recently irradiated" fuel is defined in the applicable TS Bases as "fuel that has occupied part of a critical reactor core within the previous 100 hours." The purpose of the addition of the term "recently irradiated" throughout the TSs is to establish a point where operability of those systems typically used to mitigate the consequences of an FHA is no longer required to meet the radiation exposure limits of 10 CFR 50.67. This amendment revises the TSs to eliminate TS controls

over the integrity of the fuel building and the reactor containment building and the operability of the associated building's ventilation/filtration systems after the decay period of 100 hours.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on June 4, 2001 (66 FR 30026). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendment dated March 19, 2001 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML010810433), as supplemented by letters dated July 6 (ADAMS Accession No. ML011980423), August 8 (ADAMS Accession No. ML012260302), and August 23, 2001 (ADAMS Accession No. ML012420089), (2) Amendment Nos. and to License Nos. DPR-66 and NPF-73, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment, dated August 17, 2001 (ADAMS Accession No. ML012210436). Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the ADAMS Public Electronic Reading Room on the internet at the NRC Web site,

http://www.nrc.gov/NRC/ADAMS/index.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, or 301-415-4737, or by e-mail at pdr@nrc.gov.

Dated at Rockville, Maryland, this 30th day of August 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate I
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