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Mr. Lew W. Myers
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

SUBJECT: AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. NPF-58 - PERRY NUCLEAR POWER PLANT, UNIT 1 (TAC NO. M96931)

Dear Mr. Myers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 103 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit 1. This amendment revises the Technical Specifications in response to your application dated August 27, 1996 (PY-CEI/NRR-2076L), as supplemented by submittals dated April 9, 1997 (PY-CEI/NRR-2162L), July 22, 1998 (PY-CEI/NRR-2299L), December 3, 1998 (PY-CEI/NRR-2340L), and January 18, 1999 (PY-CEI/NRR-2359L).

This amendment revised Technical Specification 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," and 3.6.1.9, "Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)." The amendment reflects implementation of the revised accident source term in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" and permits the licensee to eliminate the MSIV LCS and increase the allowable leak rates of the MSIVs.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by:

Douglas V. Pickett, Senior Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No 103 to License No. NPF-58
2. Safety Evaluation

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* See TEssig to SRichards memo of 2/18/99

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 26, 1999

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Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

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Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 103 to
License No. NPF-58
2. Safety Evaluation

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cc:

Mary E. O'Reilly
FirstEnergy Corporation
76 South Main St.
Akron, OH 44308

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
P.O. Box 331
Perry, OH 44081-0331

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4531

Sue Hiatt
OCRE Interim Representative
8275 Munson
Mentor, OH 44060

Henry L. Hegrat
Regulatory Affairs Manager
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P.O. Box 97, A210
Perry, OH 44081

William R. Kanda, Jr., Plant Manager
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P.O. Box 97, SB306
Perry, OH 44081

Mayor, Village of North Perry
North Perry Village Hall
4778 Lockwood Road
North Perry Village, OH 44081

Donna Owens, Director
Ohio Department of Commerce
Division of Industrial Compliance
Bureau of Operations & Maintenance
6606 Tussing Road
P. O. Box 4009
Reynoldsburg, OH 43068-9009

James R. Williams
Chief of Staff
Ohio Emergency Management Agency
2855 West Dublin Granville Road
Columbus, OH 43235-7150

Mayor, Village of Perry
P.O. Box 100
Perry, OH 44081-0100

Radiological Health Program
Ohio Department of Health
P.O. Box 118
Columbus, OH 43266-0118

Ohio Environmental Protection
Agency
DERR--Compliance Unit
ATTN: Mr. Zack A. Clayton
P.O. Box 1049
Columbus, OH 43266-0149

Chairman
Perry Township Board of Trustees
3750 Center Road, Box 65
Perry, OH 44081

State of Ohio
Public Utilities Commission
East Broad Street
Columbus, OH 43266-0573

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 103
License No. NPF-58

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

   A. The application for amendment by the FirstEnergy Nuclear Operating Company (the licensee, formerly The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, and Toledo Edison Company) dated August 27, 1996, as supplemented by submittals dated April 9, 1997, July 22, 1998, December 3, 1998, and January 18, 1999, 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, 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-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 103are hereby incorporated into this license. The FirstEnergy Nuclear Operating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett

Douglas V. Pickett, Senior Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 26, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 103

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.6-18	3.6-18
3.6-27	3.6-27
3.6-28	3.6-28

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.9	-------------------NOTES------------------- 1. Only required to be met in MODES 1, 2, and 3. 2. Main Steam Line leakage is not included. --- Verify the combined leakage rate for all secondary containment bypass leakage paths is $\leq 0.0504\ L_a$ when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.10	-------------------NOTE------------------- Only required to be met in MODES 1, 2, and 3. --- Verify leakage rate through each main steam line is ≤ 100 scfh when tested at $\geq P_a$, and the total leakage rate through all four main steam lines is ≤ 250 scfh, when tested at $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.1.9 Main Steam Shutoff Valves

LCO 3.6.1.9 The Main Steam Shutoff Valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each penetration flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Main Steam Shutoff Valves inoperable.	A.1 Close the inoperable Main Steam Shutoff Valve.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.9.1	Verify the isolation time of each valve is within limits.	In accordance with the Inservice Testing Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

PERRY NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated August 27, 1996, as supplemented by submittals dated April 9, 1997, July 22, 1998, December 3, 1998, and January 18, 1999, FirstEnergy Nuclear Operating Company (the licensee) requested an amendment to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No 1. The proposed amendment would (1) increase the allowable main steam line leakage through each main steam line to less than or equal to 100 standard cubic feet per hour (scfh) from 25 scfh and the combined leakage rate through the four main steamlines to less than or equal to 250 scfh from 100 scfh and (2) eliminate the main steam isolation valve (MSIV) leakage control system.

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

Specifically, the licensee requested that:

1. TS Surveillance Requirement Section 3.6.1.3.9 be amended to exempt main steam line leakage from the overall combined leakage rate for all secondary containment bypass leakage paths by adding a note that states "main steam line leakage is not included." This statement clarifies that main steam line leakage is not "double counted" since main steam line leakage is separately addressed in Bases Section SR 3.6.1.3.9.

2. Technical Specification (TS) Surveillance Requirement (SR) Section 3.6.1.3.10 be amended to limit the leakage through each main steam line to less than or equal to 100 scfh and to limit the combined leakage rate through the four main steam lines to less than or equal to 250 scfh. In a proposed revision to Bases Section SR 3.6.1.3.10, the licensee committed to restoring the leakage rate to less than or equal to 25 scfh on a main steam line if the leakage rate on that main steam line exceeds 100 scfh during the leakage testing required by the TS.

3. TS Section 3.6.1.9 be amended to permit the deletion of the MSIV Leakage Control System from the TSs. In its place, a new TS entitled "Main Steam Shutoff Valves" is to be added with operability and surveillance requirements.

In addition, the licensee also proposed conforming changes to the TS Bases and the Table of Contents to reflect the requested changes and submitted them "for information only." The TS Bases are not part of the Perry TS; therefore, these conforming changes are not subject to the staff's approval. The staff did review the proposed conforming changes to the TS Bases and concludes that the changes are consistent with the TS changes. These changes will be accomplished under the Perry Nuclear Power Plant Bases Control Program in accordance with Perry TS Section 5.5.11.

The requested amendment is based on the implementation of the revised accident source term provided in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 1). In its radiological consequence evaluation, the licensee used, and the staff approves of its use, the fission product releases during reactor coolant, gap, and early in-vessel release phases (excluding ex-vessel and late in-vessel releases) as provided in NUREG-1465.

The staff has concluded that these source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel; therefore, they are appropriate for evaluating the radiological consequences of a design basis accident. The licensee submitted this amendment with the endorsement of the Nuclear Energy Institute as the lead pilot plant application for implementing an alternative source term at operating nuclear power plants.

2.0 BACKGROUND

10 CFR Part 100 requires that a fission product release into containment be postulated and that offsite radiological consequences be evaluated against the guideline dose values specified in that regulation. The fission product releases into containment are used for evaluating the acceptability of both the plant site and the effectiveness of engineered safety feature (ESF) components and systems. The current source term was published in 1962 by the U.S. Atomic Energy Commission in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors" (TID source term).

Since that time, there have been significant advances in the staff's understanding of the timing, magnitude, and chemical forms of fission product releases from severe reactor accidents. NUREG-1465, which was published in February 1995, reflects the extensive research and experience that culminated in the development of the revised accident source term. The development of the revised accident source term was originally intended for application to advanced reactor designs and was used in the CE System 80+ and AP600 design certifications.

In SECY-96-242, "Use of the NUREG-1465 Source Term at Operating Reactors," dated November 25, 1996, the staff informed the Commission of its approach to allow the use of the revised accident source term described in NUREG-1465 at operating plants. In the SECY paper, the staff described its plans to (1) undertake a rebaselining assessment of two nuclear power plants to further evaluate the issues involved with applying the revised accident source term at operating plants, (2) review the pilot plant applications implementing the revised accident source terms following completion of the rebaselining, and (3) incorporate the total effective dose equivalent (TEDE) methodology in the review of the pilot plant applications. The

Commission approved these plans and directed the staff to commence rulemaking upon completion of the rebaselining and concurrent with the pilot plant reviews.

The staff has completed its rebaselining effort and the results are provided in SECY-98-154 (Reference 2). The staff submitted a proposed rule to amend 10 CFR Parts 21, 50, and 54 for the use of alternative source terms at operating reactors in SECY-98-289 (Reference 3) and placed a copy of the proposed rulemaking package in the Public Docket Room on October 19, 1998. Meanwhile, the staff has initiated review of license amendments submitted by pilot plants. In addition to Perry, the pilot plants that submitted license amendments using the alternate accident source term are Browns Ferry Units 1, 2, and 3, Indian Point Unit No. 2, Oyster Creek, and Grand Gulf.

3.0 EVALUATION

To demonstrate the adequacy of the Perry engineered safety features designed to mitigate the radiological consequences of the design basis accidents (DBAs) with the increased MSIV leak rate of 250 scfh and without relying upon the MSIV Leakage Control System, the licensee re-evaluated the offsite and control room radiological consequences resulting from a postulated loss-of-coolant accident (LOCA). The licensee submitted the results of its offsite and control room dose calculations in the submittals, and they are shown in Table 1.

In its submittals, the Perry licensee concluded that the existing engineered safety feature systems in combination with the increased MSIV leak rate of 250 scfh and without relying upon the MSIV Leakage Control System (LCS), will provide assurance that the radiological consequences at the exclusion area boundary (EAB) and the low population zone (LPZ) resulting from the most limiting loss-of-coolant accident (LOCA) will be within the dose criteria (25 rem TEDE) provided in the proposed rule to amend 10 CFR Part 50 (Reference 3).

The TEDE criteria, which are needed to support alternative accident source term applications, are not currently provided in regulations governing operating reactors. A detailed rationale for the use of 25 rem TEDE as an accident dose criterion and the use of the 2-hour exposure period resulting in the maximum dose for future light-water reactors (LWRs) is provided at 61 FR 65157. The same considerations that formed the basis for that rationale are similarly applicable to operating reactors such as Perry.

The licensee reached the above conclusion for meeting the proposed dose criteria by:

(1) using the reactor accident source term provided in NUREG-1465,
(2) relying on natural deposition of fission product aerosol in the drywell,
(3) relying on natural deposition of fission product aerosol in the main steam lines,
(4) controlling the pH of the water in the containment to prevent iodine evolution,
(5) operating the containment spray system for up to 24 hours,
(6) not crediting iodine removal by charcoal adsorbers in the annulus exhaust gas treatment system (AEGTS),
(7) delaying actuation of control room emergency filtration system for 30 minutes,

(8) decreasing elemental and organic iodine removal efficiencies of control room emergency filtration system charcoal adsorbers from 95 percent to 50 percent,
(9) increasing the engineered safety feature system leakage outside primary containment assumed in the Perry USAR by 50 percent, and
(10) increasing the maximum allowable containment bypass leakage in the Perry TS by 50 percent.

To review the licensee's radiological consequence assessments, the staff performed a confirmatory radiological consequence calculation for the following four potential fission product release pathways following the postulated LOCA:

(1) main steam isolation valve leakage,
(2) containment leakage,
(3) containment bypass leakage, and
(4) post-LOCA leakage from engineered safety features systems outside containment.

The fission product transport models used by the staff to calculate radiological consequences are shown in Figures 1 and 2. The results of the staff's independent radiological consequence calculation are provided in Table 1, along with those results calculated and provided by the licensee. The major parameters and assumptions used by the staff in the radiological consequence calculations are listed in Tables 2 through 8. The staff used its newly developed RADTRAD (Radionuclide Transport and Removal and Dose Estimation) computer code to calculate the resulting radiological consequences at the EAB, LPZ, and in the main control room. The RADTRAD code calculates fission product transport and removal along with the resulting radiation doses at selected receptors. This code is described in NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation" (Reference 4).

3.1 Main Steam Isolation Valve Leakage Pathway

There are four main steam lines at the Perry Nuclear Power Plant; each line has an inboard MSIV, an outboard MSIV, and a third isolation valve. These valves isolate the reactor coolant system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or other events requiring containment isolation. These MSIVs along with the main steam lines, up to and including the third isolation valve, are designed as Seismic Category 1. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage occurs through these valves. The current Perry TS limit for MSIV leakage is 25 scfh for any single MSIV. Operating experience at various boiling-water reactor (BWR) plants has indicated that degradation has occurred in the leak-tightness of MSIVs, and this specified low leakage has proven to be difficult to maintain.

Because of recurring problems with excessive leakage of MSIVs, Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," recommended the installation of a supplemental main steam leakage control system to ensure that the isolation function of the MSIVs complies with the specified limits. To meet this requirement, the licensee installed a safety-related MSIV LCS which is designed to control and minimize the release of fission products through the closed MSIVs. Without the LCS, this leakage would bypass the containment and would be released without filtration following the postulated LOCA. The LCS collects leakage from the closed MSIVs and passes the leakage flow into the annulus building served by the AEGTS.

In response to the MSIV leakage concerns, the BWR Owners Group (BWROG) commissioned a program of studies in 1986 to determine the causes of high leak rates and the means to eliminate them. The results of these studies were submitted to the NRC in General Electric proprietary reports NEDO-31643P (November 1988); NEDO-31858P, Revision 0 (February 1991); NEDO-31858P, Revision 1 (October 1991); and NEDO-31858P, Revision 2 (September 1993), all titled "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems."

These reports conclude that the increase of the MSIV leakage limit will reduce radiation exposures to maintenance personnel, reduce outage durations, and extend the effective service life of the MSIVs. The reports also conclude that the elimination of the LCS will similarly reduce exposures to maintenance personnel and reduce outage durations, and that the LCS can be replaced with an alternate method for MSIV leakage treatment utilizing the main steam lines and condenser. The licensee referenced these reports as a basis for deleting the TS requirements for the MSIV LCS and to increase the higher MSIV leakage limit.

The staff assumed in its evaluation that the fission product leakage in the main steam lines allowed by the TS is released directly into the environment. The staff provided no credit for fission product deposition and/or holdup (for decay) in the main steam lines beyond the second MSIV or in the main condenser since the licensee has not seismically analyzed them to be qualified as a holdup volume for fission products following the postulated LOCA. Sections III(c) and VI of Appendix A to 10 CFR Part 100 require that structures, systems, and components necessary to ensure the capability of mitigating the radiological consequences of an accident that could result in exposures comparable to the dose guidelines of Part 100 be designed to remain functional during and after a safe-shutdown earthquake.

The staff assumed a double guillotine pipe rupture in one of the four main steam lines upstream of the inboard MSIV and failure of all four third main steam isolation valves to close as a result of a common power failure (single-failure criterion). A total of a 250 scfh maximum allowable leakage limit is assumed to occur: (1) 100 scfh through the broken steam line, (2) 100 scfh through a second intact steam line, and (3) the remaining 50 scfh through a third intact steam line. This pathway is the major radiological consequence contributor resulting in a TEDE of 20 rem at the EAB, or approximately 89 percent of any 2-hour EAB dose.

3.1.1 Fission Product Transport in Drywell

In its evaluation, the licensee assumed that a large-break LOCA, as a result of a double guillotine pipe rupture in one of the four main steam lines upstream of the inboard MSIV, would be the most limiting DBA with respect to the offsite and control room radiological consequences. The licensee further assumed that all fission products are released directly to the drywell and leaked into the primary containment and into the main steam lines, bypassing the suppression pool. The staff concludes that this assumption is appropriate for the large-break LOCA. For small-break LOCAs with operator actuation of an automatic depressurization system (ADS), most of the fission products would be released both into the drywell through the pipe break and into the suppression pool through the ADS. Historically for the TID source term, the staff assumed that the fission products released from the reactor coolant system into the drywell are instantaneously and uniformly distributed in the drywell and in the containment at the time of the accident. Therefore, credit was allowed for suppression pool scrubbing of the TID source term in the initial blowdown.

The staff assumed that the fission products are homogeneously distributed between the drywell and the primary containment two hours after accident initiation. The objective of this well mixed approach is to achieve an appropriate balance for the design of drywell leakage mitigative devices such as the MSIVs as well as containment leakage mitigative features such as the annulus effluent gas treatment system. An overly conservative drywell leakage will result in MSIVs with lower allowable MSIV leakage but will diminish the importance of containment mitigative features and vice versa.

As characterized in NUREG-1465, the gap and early in-vessel fission product releases terminate 2 hours after accident initiation. This would require reflooding of the reactor vessel. Instead of trying to justify an all encompassing steaming rate due to this reflooding, the staff concludes that a substantial amount of fission products may end up in the primary containment as well as the drywell and that mitigative features such as the annulus effluent gas treatment system need to be designed to accommodate a significant portion of the source term. For most of the risk significant cases, such as station black out and transients, all the fission products are released directly to the primary containment via the safety relief valves. Waiting 2 hours to homogeneously mix the source term is acceptable for achieving an appropriate balance because the worst 2 hours are considered as opposed to simply the first 2 hours when using the TID source term.

Confirmatory calculations performed by the staff showed that the resulting radiological consequences are dependent upon the drywell bypass leakage prior to the termination of fission product release at 2 hours. Because of this sensitivity, the staff concludes the steaming rate of an intact core without relocation to the lower head, on the order of 3,000 cfm, should be assumed for drywell bypass leakage instead of the 6,200 cfm originally proposed by the licensee. The staff's steaming rate prior to 2 hours is conservative in that it does not credit steaming due to relocation, cooling from alternative water sources, or the release of hydrogen gas.

The 3,000 cfm drywell bypass leakage rate is based upon large-break LOCA analyses performed with MELCOR on a Grand Gulf type model (Reference 5). These analyses showed no relocation below the core plate, water level below the core plate, and an average steaming rate of approximately 2,800 cfm prior to quenching of the core at approximately 0.5 hours. Also, alternative water sources, such as the standby liquid control system, would not be available during station blackout sequences which comprise 96% of the core damage frequency for Grand Gulf according to NUREG-1150. Therefore, the staff concludes the use of 3,000 cfm for the drywell bypass leakage prior to 2 hours is reasonable for Perry.

By letter, dated January 18, 1999, the licensee for Perry supplemented the license amendment request. Attachment 1 to the supplement stated that the sweepout flow rate from drywell to containment during the fission product release phase is assumed to be 3,000 cfm, instead of 6,200 cfm as originally proposed in the supporting dose calculations. Also, the supporting dose calculations now assume that the drywell and the unsprayed region of the containment are well-mixed after 2 hours. Based on the discussion above, the staff finds these revised assumptions, as described in the January 18, 1999 letter, acceptable.

3.1.2 Aerosol Deposition Within The Drywell

In its evaluation, the staff used a simplified model developed by the staff for estimating the fission product aerosol deposition by natural processes in the drywell of BWRs following a postulated LOCA. The model is described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containment" (Reference 6). This model was derived by correlating the results of Monte Carlo uncertainty sampling analyses assessing uncertainties in aerosol properties, drywell geometries, accident progression, and aerosol behavior expected to be associated with a postulated LOCA in the drywell.

The staff considered that the fission product aerosols in the drywell are removed by natural processes: gravitational sedimentation and phoretic phenomena such as diffusiophoresis and thermophoresis. The staff assumed that the drywell is well mixed during the entire duration of the accident. The aerosol removal rates used by the staff represent the 90th percentile of the uncertainty distributions.

3.1.3 Aerosol Deposition in The Main Steam Lines

The Perry main steam lines consist of four 24-inch-diameter carbon steel pipes. These lines are welded to the reactor vessel nozzles and run parallel to the vertical axis of the vessel, downward to the elevation where they emerge horizontally from the drywell. Three MSIVs are installed on each steam line: one inboard and one outboard of the drywell, and one inboard of the reactor building as a third isolation valve. The length of each steam line from the inboard MSIV to outboard MSIV is approximately 49 feet, and from the outboard MSIV to the third MSIV is 29 feet.

The staff model treats the MSIV leakage pathway as three segments for which instantaneous and homogeneous mixing is assumed. The first segment is the length of intact piping between the reactor vessel and the first MSIV. The second segment is the length of intact piping between the first MSIV and the second MSIV. The third segment is the broken main steam line between the first MSIV and the second MSIV. The staff considered only gravitational sedimentation as the aerosol removal mechanism in the main steam lines.

During the postulated LOCA, the main steam leakage flow pattern in the main steam lines could be plug flow, well-mixed flow, or some combination of the two. If temperature gradients exist along the length of the main steam pipe, some degree of mixing would occur. For the same leak rate into the main steam line, plug flow is expected to result in less offsite release than well-mixed flow since the concentration of the fission product released to the environment is at the concentration of the fission product in the plug at the end of the main steam pipe. Plug flow effectively results in a longer fission product transport time in the pipe and, therefore, more deposition in the pipe.

The licensee has performed an analysis to show that flow in some parts of the main steam line is plug flow. The staff believes that current modeling uncertainties are such that it is problematic to treat these regions as not undergoing some degree of mixing. The staff, therefore, concludes that, at this time, a well-mixed model is more appropriate than a plug flow model for settling in the main steam line. However, complete mixing may not occur along the entire length of the pipe and, in some pipe segments, plug flow may exist. For its analysis, the staff has chosen to represent the flow in the main steam line as well-mixed.

The staff has also chosen a conservative settling rate (Table 2) to assess the fission product deposition. This conservative settling rate was determined based on a Monte Carlo analysis of settling velocity using the ranges and distributions given in References 6 and 7 for the uncertain parameters. The staff's analysis includes additional conservatism because no credit was provided for additional deposition by thermophoresis, diffusiophoresis, and flow irregularities; additional deposition as a result of hygroscopicity; possible plugging of the leaking MSIV by aerosols; and additional deposition in the piping between the outboard MSIV and the third isolation valve.

3.2 Containment Leakage Pathway

The primary containment consists of a drywell, a wetwell, and supporting systems to limit fission product leakage during and following the postulated LOCA with rapid isolation of the containment boundary penetrations. The design basis leak rate of the primary containment is 0.2 volume percent per day. The staff assumed the flow rates given in Table 3 which are based on this design basis leak rate for the entire duration of the accident (30 days).

The secondary containment (shield building) which surrounds the primary containment will collect and retain any fission product leakage from the primary containment and will release fission products to the environment in a controlled manner through the AEGTS. During normal plant operation, the shield building is maintained at a slight negative pressure at a vacuum of 0.4-inch water gauge. The Perry updated safety analysis report (UFSAR) states that the secondary containment pressure is expected to remain negative following a DBA. However, for a short period, it may not be maintained below the negative pressure of 0.25-inch water gauge. Therefore, the licensee assumed, and the staff agrees, that the entire primary containment leakage is released directly to the environment for the first 40 seconds into the postulated LOCA.

Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary containment and the shield building boundaries that could create potential pathways through which fission product in the primary containment could bypass the leakage collection and filtration systems associated with the shield building. The Perry Technical Specifications limit the secondary containment bypass leakage to equal to or less than 5.04 percent of the primary containment leak rate. The licensee proposed to use 10 percent of the primary containment leak rate in this evaluation to be more conservative.

Therefore, the staff used 90 percent of the primary containment leak rate into the secondary containment following the 40 seconds for its radiological consequence analysis. This leakage is collected in the shield building and processed through the AEGTS before being released into the environment. This pathway contributed a TEDE of 0.6 rem at the EAB, or approximately 2.7 percent of the 2-hour EAB dose. The remaining 10 percent of the primary containment leak rate is assumed to have bypassed the shield building and to have been released from the primary containment directly to the environment for the entire duration of the postulated LOCA. This pathway resulted in a TEDE of 1.4 rem at the EAB, or approximately 6 percent of the 2-hour EAB dose.

3.2.1 Annulus Effluent Gas Treatment System

The AEGTS is an engineered safety features system and is designed to collect, process, and release the fission product leakage from the primary containment into the shield building. The AEGTS is a redundant system consisting of two 100 percent capacity subsystems. Each subsystem has a design capacity of 2000 cfm and consists of, among other things, a pre-HEPA filter, one 4-inch deep charcoal adsorber, and a post-HEPA filter. The system is designed to Seismic Category 1 standards and is located in a Seismic Category 1 structure.

The system is operated continuously during normal plant operation, and it maintains a slight negative pressure in the shield building. The staff assumed a 99 percent removal efficiency for fission products in aerosol form for HEPA filters. The licensee has not requested, and the staff has not provided, any iodine removal efficiency for the charcoal adsorbers in the AEGTS in its respective radiological consequence calculations. HEPA filters are periodically tested in accordance with the guidelines provided in Regulatory Guide 1.52.

3.2.2 Containment Spray

The containment spray system (CSS) is an engineered safety features system and is designed to provide containment cooling and fission product removal in the containment following a postulated LOCA. The CSS consists of two redundant and independent loops. Each loop has a design spray water flow capacity of 5250 gpm. The system is designed to Seismic Category 1 standards and is located in a Seismic Category 1 structure. No chemical additives are used in the CSS other than pH buffering chemical (sodium pentaborate) in the existing Standby Liquid Control System (see Section 3.4).

The licensee assumed a mixing rate of 6.3 unsprayed volumes per hour between the sprayed and unsprayed portions of the containment atmosphere by operation of the CSS. The proposed mixing rate is higher than the two turnovers of the unsprayed region per hour specified in the Standard Review Plan Section 6.5.2. The staff accepted the proposed mixing rate on the basis of its review of the licensee's calculation that demonstrated that an adequate mixing flow existed between unsprayed and sprayed regions by natural convection. The difference in these two mixing rates resulted in less than a TEDE of 0.3 rem at the EAB (less than 1.3 percent of the 2-hour EAB dose).

In the letters dated August 27, 1996, and July 22, 1998, the licensee committed that the CSS would be operated post-LOCA for up to 24 hours (the current analyses in the Perry USAR assumed 6 hours of spray operation). The licensee stated that work has already progressed on additional plant operator guidance for events that might result in large release of fission product such as the revised accident source term. The licensee's new guidelines direct that the CSS be initiated manually based on readings from the containment high range radiation monitor. Otherwise, the CSS will automatically initiate 10 minutes following a LOCA signal if containment pressure exceeds the high pressure setpoint. This work has been done by the licensee under the Severe Accident Management (SAM) mitigation effort. The SAM guidelines became effective in December 1998. In its dose calculation, the staff assumed the CSS are operated for the first 24 hours.

In its evaluation, the staff used a simplified model for estimating the fission product aerosol removal by containment sprays following a postulated LOCA. The model is described in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays" (Reference 7). This model was derived by correlating the results of Monte Carlo uncertainty sampling analyses assessing the uncertainties in aerosol properties, aerosol behavior, spray droplet behavior, and the initial and boundary conditions expected to be associated with a postulated LOCA in the containment. Two parameters used in this evaluation that are not treated as uncertainty distributions for Perry are (1) spray water flux ($0.0621 ft^3 / ft^2$ min), and (2) mean spray fall height (53.2 ft). These parameters were specified based on plant specific design information for Perry. The staff used 90th percentile uncertainty distributions for fission product in aerosol form in its radiological consequence calculations. Spray removal rates for elemental iodine in the sprayed region are listed in Table 3.

3.3 Post-LOCA Leakage Pathway From Engineered Safety Features Outside Containment

Any leakage water from ESF components located outside the primary containment releases fission products during the recirculation phase of long-term core cooling following a postulated LOCA. In the Perry UFSAR, the licensee estimated this leakage to be less than 5 gallons per hour (gph) and used 10 gph for its radiological consequence calculations in Chapter 15 of the UFSAR for conservatism. In its license amendment request, the licensee proposed, and the staff accepted, the use of 15 gph of ESF leakage for the entire duration of the accident (30 days). Additionally, leakage from a gross failure of a passive component is assumed to occur at a rate of 50 gpm starting 24 hours into the accident and lasting for 30 minutes. Ten percent of iodine (all forms) contained in the leakage was assumed to be released directly to the environment and the pH of water leakage was assumed to be above 7 (see Section 3.4). This pathway resulted in a TEDE of 0.5 rem at the EAB, or approximately 2.4 percent of the 2-hour EAB dose.

3.4 Post-Accident Containment Water Chemistry Management

In NUREG-1465, the staff concluded that iodine entering the containment from the reactor coolant system during an accident would be composed of at least 95 percent of cesium iodide (CsI) with no more than 5 percent of iodine (I) and hydrogen iodide (HI). Once in the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide (I^-) in solution. The staff also stated that the radiation-induced conversion of iodide in water into elemental iodine (I_2) is strongly dependent on the pH. The staff indicated that without pH control, a large fraction of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be released into the containment atmosphere if the pH is less than 7. On the other hand, if the pH is maintained above 7, very little (less than 1 percent) of the dissolved iodine will be converted to elemental iodine.

The licensee proposed to use the existing Standby Liquid Control System (SLCS) for controlling and maintaining long-term suppression pool water pH levels to 7 or above following the postulated DBA. The SLCS is a safety-related system and designed as a Seismic Category 1 system. It is designed as a reactivity control system and provides backup capability so as to be able to shut down the reactor if the normal control becomes inoperable. The Perry TS requires the system to be maintained in an operable status whenever the reactor is critical. The system is manually initiated from the main control room to pump a boron neutron absorber solution into the reactor in accordance with the Perry Plant Emergency Instructions.

The SLCS contains 5,236 lbs of sodium pentaborate ($Na_2B_{10}O_{16}10H_2O$). Solutions of sodium pentaborate act as pH buffers. Buffering will cause only a small decrease in pH with addition of an acid so long as the buffer capacity is not exceeded. The licensee used a containment water pool volume (which includes the suppression pool and reactor coolant inventory) of 1.3E+6 gallons and assumed all cesium iodide released into the drywell is directly deposited in the containment water pool. In the analysis of pH levels in the containment water pool, the licensee considered the following factors:

(1) the addition of sodium pentaborate
(2) hydrochloric acid generated from electrical cable degradation
(3) cesium hydroxide formed from the fission products released from the core
(4) nitric acid produced by irradiation of water and air in the containment

The licensee used the methods and models described in NUREG/CR-5950, "Iodine Evolution and pH Control," to determine the formation of hydrochloric and nitric acids. The licensee concluded that with the amount of sodium pentaborate provided in the containment, the pH of the post-accident water in the containment will remain above 7 for the entire duration of the postulated LOCA.

To verify the licensee's conclusion, the staff (through its contractor) experimentally measured the pH of a solution composed of the same proportions of containment water, sodium pentaborate, and cesium hydroxide expected to be present in the Perry containment following the postulated LOCA. The initial pH measured was 8.6. This value is consistent with the licensee's calculated value. After this initial pH measurement, the solution was titrated with nitric acid to simulate the radiolytic production of hydrochloric and nitric acid formation from water, air, and electric cables.

The total amount of acid produced is calculated to be approximately 477 moles for the 30-day accident period, while the licensee calculated it to be 400 moles. The staff measured and calculated the pH values to be 8.2 and 8.5 at 30 days, with 477 moles of acid. The licensee's calculated pH value at 30 days is 8.0 with 400 moles of acid. The staff concludes the licensee's calculated pH values are acceptable. The staff's calculated and measured pH values are shown in Table 8 and in Figure 3, respectively.

In its evaluation, the staff assumed that the containment water pool is well mixed with sodium pentaborate solution. The staff concludes that for the first 2 hours into a DBA, the iodide source term behavior and its transport within the drywell will be independent of iodine re-evolution and pH control. Therefore, the staff further concludes that the manual initiation feature of the SLCS and the well mixed assumptions are acceptable. The sodium pentaborate solution will be well mixed with the containment water pool by the end of the 2-hour period as a result of reflood of the reactor vessel.

Based on the above evaluation, the staff concludes that the licensee's calculated pH values of the containment water pool following the postulated LOCA are acceptable and that the existing SLCS design is capable of controlling and maintaining long-term suppression pool water pH levels at 7 or above during the entire 30 day-period of the accident.

3.5 Control Room Habitability

Upon receipt of an ESF actuation system signal or high radiation, the Perry control room heating, ventilation, and air conditioning (HVAC) system is designed to automatically switch to the emergency recirculating mode of operation, activating the control room emergency recirculation system (CRERS). The licensee proposed, and the staff used in its evaluation, a 30-minute delay in actuation of the CRERS.

The CRERS is a redundant system and each subsystem has a design flow capacity of 30,000 cfm. The licensee proposed, and the staff used in its evaluation, a conservative recirculation flow rate of 27,000 cfm. Each subsystem consists of, among other things, a high-efficiency particulate air filter, charcoal adsorbers, and a post HEPA filter. The licensee proposed, and the staff used in its evaluation, a HEPA filter efficiency of 95 percent for aerosol particulate and a 50 percent charcoal filter removal efficiency for iodine in elemental and organic forms.

The HVAC system is designed to pressurize the control room envelope with 45,000 cfm recirculation air flow and with 6,000 cfm outside makeup air during normal plant operation. During an emergency, when the system operates in the emergency recirculation mode, the outside makeup air is isolated and the control room envelope is not pressurized relative to adjacent areas. To be conservative, the licensee proposed to use 1375 cfm inleakage to the control room during the emergency recirculation mode for the entire duration of the accident as stated in Section 6.4 of the Perry USAR. The major parameters and assumptions used by the staff are listed in Table 7. The staff's independently calculated TEDE for control room personnel following the postulated LOCA for the entire duration of the accident is 5 rem, which is same as that calculated by the licensee.

By letter dated December 3, 1998, the licensee requested an exemption from the control room dose acceptance criterion of 10 CFR Part 50, Appendix A, General Design Criteria 19, "Control Room." The requested exemption would permit use of a 5 rem control room TEDE acceptance criteria in lieu of "5 rem whole body, or its equivalent to any part of the body," as currently stated in GDC 19. The staff is currently proposing to replace the GDC 19 dose criteria for operating reactors that elect to use an alternative source term with a TEDE criterion of 5 rem in the proposed rule, 10 CFR Part 50.67.

The staff has reviewed the requested exemption and finds it acceptable because the staff has concluded that the TEDE methodology provides an alternative means of meeting the current regulatory requirement. The staff's conclusion was based on the language in General Design Criterion (GDC) 19: "5 rem whole body, or its equivalent to any part of the body" which is subsumed by the definition of TEDE in 10 CFR 20.1003 and by the 5 rem annual limit for TEDE in 10 CFR 20.1201(a). The Commission directed the staff to consider, on a case-by-case basis, technically justified exemptions to facilitate the pilot plant review pending completion of such rulemaking.

The staff further concludes there is no significant radiological environmental impact as a result of the requested exemption and the staff's assessment was forwarded to the Office of the Federal Register for publication. The control room operator dose calculated by both the staff and the licensee are within the 5 rem control room TEDE acceptance criteria requested in the exemption and it is also within the dose criteria specified in the proposed rule to amend 10 CFR Part 50. Therefore, the staff concludes that the control room operator dose calculated by the licensee is acceptable.

3.6 Atmospheric Relative Concentrations at Control Room, Exclusion Area Boundary, and Low Population Zone

The licensee conducted an atmospheric tracer study to characterize the atmospheric dispersion within the building complex at the Perry plant. Prior estimates of atmospheric relative dispersion (X/Q) values had been made for postulated releases to the control room using the Murphy-Campe methodology referenced in Standard Review Plan 6.4. The primary objective of conducting the tracer tests was to demonstrate that a reduction in the magnitude of the control room air intake X/Q values was appropriate. The tests were conducted during a one-week period in September 1985.

The NRC reviewed and compared the results of the tracer study with calculations made using the ARCON96 methodology described in NUREG/CR-6631, Rev.1, "Atmospheric Relative Concentrations in Building Wakes" (Reference 8). The licensee provided measurements of hourly meteorological data collected at the Perry site from calendar years 1993 through 1997 which the staff used as input to the assessment.

For the postulated release point resulting in the largest X/Q values, the calculated X/Q values from the tracer study were as much as 50 times lower than the original X/Q values calculated by the licensee using the Murphy-Campe methodology. For the same postulated release point, X/Q values using the ARCON96 methodology were as much as 25 times higher than those calculated from the field tests.

ARCON96 was developed to provide better description of atmospheric dispersion near buildings than is provided by the Murphy-Campe methodology. ARCON96 is based upon modifying a Gaussian dispersion model to fit experimental data collected in field experiments at nuclear power plants. The ARCON96 methodology assumes that the effluent travels the shortest distance possible between the postulated release point and the control room air intake. While the model calculates dispersion within building complexes, it is not intended to provide an exact model of postulated scenarios for complex site-specific flow paths around obstructions.

Meander and building effects are implicitly factored in, based on the field test studies used in the development of ARCON96.

At Perry, the effluent from a release postulated from the plant vent or containment building would need to disperse over or around an obstruction, down the side of a building and around a missile shield to be drawn into the control room air intake. For the limiting case, the X/Q values calculated from field tests performed by the licensee are about a factor of two to three lower for the control room air intake than for measurements made at the top of the building on which the intake is located. Thus, it is expected that results using ARCON96 would overestimate X/Q values for this scenario at Perry.

The field tests performed by the licensee were conducted over a period of approximately one week. While care was taken to assure that the tests were made under adequately limiting meteorological conditions, there is some likelihood that testing may not have captured the full range of poor dispersion conditions. Also, the field measurements may include some off-centerline conditions, and due to solar heating in the building complex, better dispersion may have occurred during the tests than might occur at some other times of the year.

After discussing the tracer test limitations with the staff, the licensee revised the X/Q values. The results for the limiting case are as follows:

Accident Period	Licensee's Murphy-Campe X/Qs (s/m^3)	Licensee's Tracer Test X/Qs (s/m^3)	Licensee's Revised X/Qs (s/m^3)	ARCON96 X/Qs (s/m^3)
0 - 8 hours	3.5 E-3	7.0 E-5	3.5 E-4	1.8 E-3
8 - 24 hours	2.1 E-3	5.6 E-5	2.1 E-4	7.3 E-4
1 - 4 days	1.1 E-3	4.3 E-5	1.1 E-4	4.7 E-4
4-30 days	2.3 E-4	1.5 E-5	5.75E-5	2.9 E-4

The staff concludes that the licensee's revised X/Q values are acceptable for the postulated design basis accident releases from the plant vent and containment building.

The staff has also reviewed the licensee's X/Q calculations and performed confirmatory calculations for the EAB and LPZ. The licensee and NRC used similar calculational methodologies based upon the Regulatory Guide 1.145. The licensee's estimates are approximately 10 to 20 percent lower than the staff's calculations which could be due to differences in the specifics of the calculational procedures. The staff concludes that the differences are not significant.

The results are as follows:

Accident Period	Distance (m)	Licensee's X/Qs (s/m^3)	1982 SER X/Qs (s/m^3)
0 - 2 hrs	863 (EAB)	4.3 E-4	4.9 E-4
0 - 8 hrs	4002 (LPZ)	4.8 E-5	5.8 E-5
8 - 24 hrs	4002 (LPZ)	3.3 E-5	4.0 E-5
1 - 4 days	4002 (LPZ)	1.4 E-5	1.6 E-5
4-30 days	4002 (LPZ)	4.1 E-6	4.6 E-6

The staff concludes that the licensee's X/Q values are acceptable for the postulated DBA releases from the plant vent and containment building.

3.7 Equipment Qualification and Plant Shielding

In section 3.11 of the Perry USAR, the licensee evaluated the environmental qualification of mechanical and electrical equipment that is required to perform design safety functions under normal, abnormal, and accident environmental conditions. In this evaluation, the licensee used the TID source term for developing the radiation environment to qualify the safety-related equipment. To compare the radiation environment resulting from the use of the TID source term with that resulting from the use of the revised source term, the licensee performed an integrated assessment to address equipment qualification of safety-related equipment and access to vital areas including the post accident sampling system.

The licensee stated in its letter dated January 18, 1999, that the long-term (180 days) post-LOCA integrated radiation doses to the safety-related equipment and to the vital areas resulting from the revised source term are below that resulting from the TID source term and well below the equipment qualification and shielding envelopes assumed in the Perry USAR. The licensee concludes, therefore, the existing environmental qualification envelopes based on the TID source term provide adequate margin for the revised source term for equipment qualification and plant shielding.

To gain insight into the impact of implementing the revised accident source term on equipment qualification at operating reactors, the staff calculated the integrated gamma and beta doses for equipment exposed to the containment atmosphere and to the containment water pool. The staff performed these dose calculations as a part of the rebaselining assessment for use of the revised accident source term at operating reactors. The results of these calculations were compared with the results of the dose calculations with the TID source term. The staff used Grand Gulf and Surry for assessing the impact of implementing the revised accident source term on equipment qualification.

The integrated airborne gamma and beta doses with the TID source term for Grand Gulf were about the same as those with the revised source term, because only noble gases and iodine were assumed to be airborne and the magnitudes of the noble gases and iodine releases of these two source terms are about same. For the containment water pools for Grand Gulf, the gamma doses are slightly higher after approximately 150 days following a postulated LOCA for the revised source term than the TID source term. This is due to the large amounts of cesium in the revised source term. The TID source term included less than 1 percent of the core cesium, while the revised source term includes 30 percent of the cesium. More detailed assessment on this comparison is provided in SECY-98-154 (Reference 2).

Existing safety-related equipment at Perry Unit No 1 is presently qualified to the integrated doses resulting from the TID source term. Staff calculations using the revised source term indicated (in the rebaselining study of Grand Gulf) that at the containment center, the long-term integrated gamma and beta doses are essentially the same for both the TID and the revised source terms. The long-term gamma dose at the containment water pool is only slightly higher with the revised source term. The licensee stated that the TID source term used for the Perry containment water pool included 50 percent of cesium instead of less than 1 percent specified in the TID source term. However, the increase in long-term gamma dose using the revised source term still remains below the equipment qualification and shielding envelopes assumed in the Perry USAR.

The staff concludes that the use of the revised source term at Perry does not represent a significantly more severe radiation environment to the safety-related equipment. Therefore, the staff concludes that the existing environmental qualification envelopes based on the TID source term provide adequate margin for equipment qualification and plant shielding.

4.0 SUMMARY

The staff has reviewed the licensee's analysis and performed a confirmatory assessment of the radiological consequence resulting from the postulated LOCA. The calculated doses by the staff and the licensee are listed in Table 1. As shown in the table, the calculated dose by the licensee is within 10 percent of that calculated by the staff for the EAB for the worst-case 2-hour period. For the 30-day LPZ dose, the value calculated by the licensee is higher than that calculated by the staff but it is well within the acceptable dose criterion. Control room personnel dose calculated by the staff is lower than that calculated by the licensee and both calculated values are within the acceptable dose criterion. Considering the many uncertainties in the modeling of fission product transport and removal mechanisms, the staff concludes that the differences in calculated doses by the staff and the licensee are not significant. Therefore, the staff concludes that the radiological consequences analyzed and submitted by the licensee are acceptable.

On the basis of this evaluation, the staff concludes that the license amendment requested by the licensee to delete the MSIV LCS and to increase the allowable MSIV leakage is acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (63 FR 53958). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

7.0 CONCLUSION

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

Principal Contributors: Richard Emch
Jay Lee
Michael Snodderly
Leta Brown
Charles Tinkler
Jason Schaperow
Chester Gingrich
John Ridgely

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8.0. REFERENCES

(1) L. Soffer et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

(2) SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," June 30, 1998.

(3) SECY-98-289, "Proposed Amendments to 10 CFR Parts 21, 50, and 54 Regarding Use of Alternative Source Terms at Operating Reactors," December 15, 1998.

(4) S. L. Humphreys et al., "RADTRAD: A Simplified Model For RADionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, SAND98-0272, Sandia National Laboratories, Albuquerque, NM, 1998.

(5) J. J. Carbajo, "MELCOR DBA LOCA Calculations," ORNL/NRC/LTR - 97/21, Oak Ridge National Laboratory, TN, January 1999.

(6) D. A. Powers, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," NUREG/CR-6189, SAND94-0407, Sandia National Laboratories, Albuquerque, NM, July 1996.

(7) D. A. Powers and S. B. Burson, "A Simplified Model of Aerosol Removal by Containment Sprays," NUREG/CR-5966, SAND92-2689, Sandia National Laboratories, Albuquerque, NM, June 1993.

(8) J. V. Ramsdell, Jr., and C. A. Simmonen, "Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, PNNL-10521, Rev. 1, Pacific Northwest National Laboratory, May 1997.

(9) Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA, 1998.

(10) Federal Guidance Report No. 12, "External Exposure to Radionuclides In Air, Water, and Soil," EPA, 1993.

TABLE 1

Radiological Consequences Expressed as TEDE
(rem)

Pathways	EAB[1]		LPZ[2]		Control Room	
	NRC	Perry	NRC	Perry	NRC	Perry
MSIV Leak	20	[3]	4.6	[3]	1.3	[3]
Containment Leak	0.6	[3]	0.6	[3]	0.3	[3]
Containment Bypass	1.4	[3]	0.2	[3]	0.1	[3]
ECCS Leak	0.5	[3]	1.4	[3]	1.7	[3]
TOTAL	22.5	20.6	6.8	9.8	3.4	4.1
Dose Criteria[4]	25		25		5.0	

[1] Exclusion Area Boundary
[2] Low Population Zone
[3] Not provided
[4] Proposed rule to amend 10 CFR Part 50

Table 2
Parameters and Assumptions Used in
Radiological Consequence Calculations
Main Steam Isolation Valve Leakage Pathway

Parameter	Value
Reactor power	3758 MWt
Drywell volume	2.76×10^5 ft^3
Wetwell volume	1.16×10^6 ft^3
First intact main steam line volume	440 ft^3
Second intact main steam line volume	292 ft^3
Leakage rates for intact main steam line	
from drywell to first volume	
0 - 2 hours	2.98 ft^3/min
2 hours - 30 days	2.47 ft^3/min
from first volume to second volume	
0 - 30 days	4.775 ft^3/min
from second volume to environment	
0 - 30 days	4.775 ft^3/min
Volume of ruptured main steam line	146 ft^3
Leakage rates for ruptured main steam line	
from drywell to main steam line volume	
0 - 2 hours	1.987 ft^3/min
2 hours - 30 days	1.65 ft^3/min
from main steam line volume to environment	
0 - 30 days	3.183 ft^3/min
Aerosol removal rate in drywell	90 percent uncertainty distribution
Aerosol removal rate in main steam lines	6.26/hr

Table 3
Parameters and Assumptions
used in
Radiological Consequence Calculations

Containment Leakage Pathway

Parameter	Value
Reactor power	3758 MWt
Volume of sprayed region	4.81×10^5 ft^3
Volume of unsprayed region	6.84×10^5 ft^3
Flow rate from drywell to unsprayed region	
0 - 2 hours	3000 ft^3/min
2 hours - 30 days	2.76×10^5 ft^3/min
Flow rate from unsprayed region to drywell	
0 - 2 hours	0 ft^3/min
2 hours - 30 days	2.76×10^5 ft^3/min
Flow rate between drywell and sprayed region	0 ft^3/min
Flow rate from sprayed region to unsprayed region	70,000 ft^3/min
Flow rate from unsprayed region to sprayed region	70,000 ft^3/min
Containment leak rate to environment from sprayed region	
0 - 40 seconds	1.00 ft^3/min
40 seconds - 30 days	0.068 ft^3/min
Spray removal rate for particulate	90 percent uncertainty distribution
Spray water flux	0.0621 (ft^3)/(ft^2 min)
Spray fall height	53.2 ft

Table 3
(Continued)

Spray removal rate for elemental Iodine (sprayed region only)	
0 - 0.7025 hours	14.3/hr
0.7025 - 1.862 hours	3.84/hr
1.862 - 2.049 hours	6.55/hr
2.049 - 2.155 hours	3.35/hr
2.155 - 24 hours	1.20/hr
Containment leak rate to environment from unsprayed region	
0 - 40 seconds	2.00 ft^3/min
40 seconds - 30 days	0.135 ft^3/min
Containment leak rate to annulus from sprayed region	
0 - 40 seconds	0 ft^3/min
40 seconds - 30 days	0.603 ft^3/min
Containment leak rate to annulus from unsprayed region	
0 - 40 seconds	0 ft^3/min
40 seconds - 30 days	1.205 ft^3/min
Annulus volume	1.96×10^5 ft^3
Flow rate from annulus to environment	2000 ft^3/min
Annulus exhaust gas treatment system filter efficiency	
particulate	99 percent
elemental and organic iodine	0

Table 4
Parameters and Assumptions
Used in
Radiological Consequence Calculations

Emergency Core Cooling system Leakage Pathway

ECCS Leakage Model

Parameter	Value
Plant Power	3758 MWt
Release Fractions and Timing	As specified for BWR in NUREG-1465 (gap and early in-vessel iodine releases only)
Release Location	Directly to suppression pool
Suppression Pool Water Volume	1.47×10^5 ft^3
ECCS Leak Rate	
0 - 24 hours	0.033 ft^3/min
24 - 24.5 hours	6.7 ft^3/min
24.5 hours - 30 days	0.033 ft^3/min
Partition Factor	10

Table 5

Meteorological Data

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0-720	$4.3x10^{-4}$

Low Population Zone Distance

Time (hr)	X/Q (sec/m^3)
0-8	$4.8x10^{-5}$
8-24	$3.3x10^{-5}$
24-96	$1.4x10^{-5}$
96-720	$4.1x10^{-6}$

Control Room

Time (hr)	X/Q (sec/m^3)
0-8	$3.5x10^{-4}$
8-24	$2.1x10^{-4}$
24-96	$1.1x10^{-5}$
96-720	$5.8x10^{-5}$

Table 6
Breathing Rates and Other Data

Exclusion Area Boundary

Time (hr)	Breathing Rate (m^3/sec)
0-720	3.47×10^{-4}

Low Population Zone

Time (hr)	Breathing Rate (m^3/sec)
0-8	3.47×10^{-4}
8-24	1.75×10^{-4}
24-720	2.32×10^{-4}

Control Room

Time (hr)	Occupancy Factor	Effective Breathing Rate (m^3/sec)
0-24	1.0	3.47×10^{-4}
24-96	0.6	2.082×10^{-4}
96-720	0.4	1.388×10^{-4}

Fission Product Inventory in Core:	Provided by the licensee based on current licensed fuel exposure (burnup) limit
Dose Conversion Factors for Inhalation:	EPA Federal Guidance Report 11 (Reference 9)
Dose Conversion Factors for Cloudshine:	EPA Federal Guidance Report 12 (Reference 10)

Table 7

Control Room Model

Parameter	Value
Volume	3.44×10^5 ft^3
Flow Rate - Unfiltered inleakage	1375 ft^3/min
Flow Rate - Exhaust	1375 ft^3/min
Recirculation Flow Rate	
0 - 0.5 hour	0
0.5 hour - 30 days	2.7×10^4 ft^3/min
Recirculation Filter Efficiencies	
particulate	95%
elemental and organic iodine	50%

Table 8

Acid Generation in containment and Calculated pH Values in Containment Water Pool
(accumulative in moles other than calculated pH)

Hours	HCL	HNO_3	HI	CsOH	Calculated pH
1.0	8.4	2.2	2.0	407	8.61
2.0	39	8.3	4.6	868	8.63
5.0	133	26	4.6	868	8.63
12	268	54	4.6	868	8.62
24	425	91	4.6	868	8.60
72	767	187	4.6	868	8.58
240	1317	348	4.6	868	8.53
480	1623	448	4.6	868	8.50
720	1745	503	4.6	868	8.48

Figure 1
Fission Product Transport Model

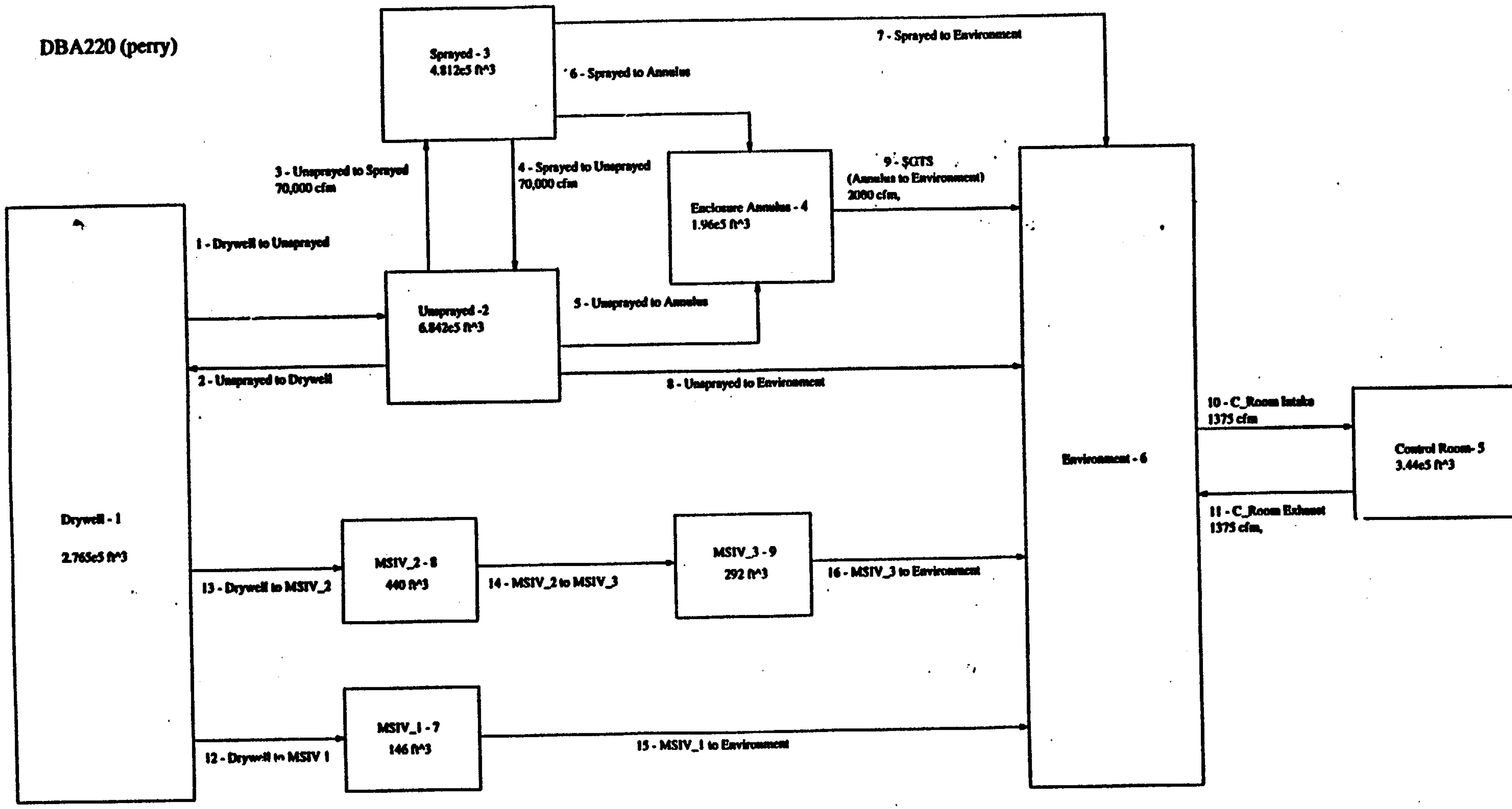

Figure 2
Fission Product Transport Model
(ECCS Leakage Pathway)

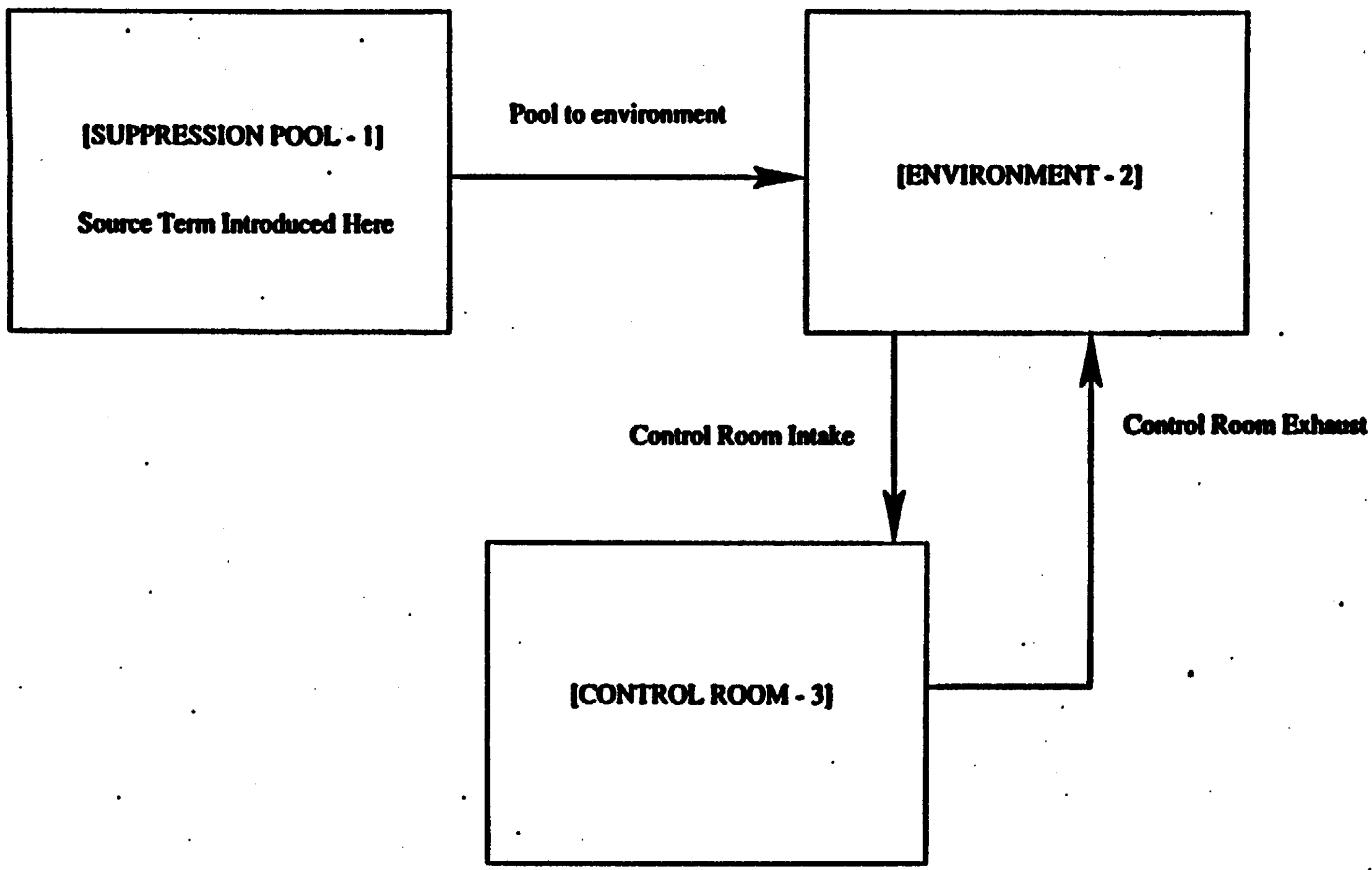

Figure 3

pH Measurements (4)