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June 5, 2002
IPN-02-044

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

SUBJECT: Indian Point Nuclear Generating Unit No.3
Docket No. 50-286
Proposed Changes to Technical Specifications:
Selective Adoption of Alternate Source Term and
Incorporation of Generic Changes; TSTF-51, TSTF-68, and TSTF-312

REFERENCES:

1. TSTF-68, Revision 2, "Containment Personnel Airlock Doors Open During Fuel Movement," NRC approved August 16, 1999.

2. TSTF-312, Revision 1, "Administratively Control Containment Penetrations," NRC approved August 16, 1999.

3. TSTF-51, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," NRC approved November 1, 1999.

4. Regulatory Guide 1.183; "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," dated July 1, 2000.

Dear Sir:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc, (ENO) hereby requests amendments to the Operating License for Indian Point 3 Nuclear Generating Unit No.3, pertaining to the implementation of the alternate source term (AST) methodology for the fuel handling accident analysis. Specifically, this license amendment request proposes the following changes to the Technical Specifications

a. Revise Technical Specification 3.9.3.a to permit the equipment hatch opening to be capable of being closed during movement of irradiated fuel.

b. Revise Technical Specification 3.9.3.b to permit the personnel air lock doors to be capable of being closed during movement of irradiated fuel. The proposed revision adopts TSTF-68 (Reference 1).

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c. Revise Technical Specification 3.9.3.c to allow the use of administrative controls for unisolating containment penetrations during movement of irradiated fuel. The proposed revision adopts TSTF-312 (Reference 2).

d. Delete Technical Specification 3.9.3.d and 3.9.3.e (and associated surveillances) regarding containment purge and containment pressure relief requirements with the reactor subcritical for less than 550 hours. Application of the AST methodology to the fuel handling accident eliminates the need for this plant specific requirement previously established by Amendment 175 to the Technical Specifications.

e. Revise the applicability of Technical Specification 3.9.3 to eliminate 'during core alterations'. The proposed revision adopts a portion of TSTF-51 (Reference 3). ENO may propose to adopt the other portions of TSTF-51 in a future license amendment request.

f. Relocate the requirements for the fuel storage building emergency ventilation system and associated actuation instrumentation (Technical Specifications 3.7.13 and 3.3.8, respectively) to the Technical Requirements Manual which is a licensee-controlled document subject to the requirements of 10 CFR 50.59.

The Indian Point 3 licensing basis for the fuel handling accident radiological consequences analysis is currently based on the methods and assumptions of TID-14844 as cited in 10CFR100. This application for amendment proposes to adopt an alternate source term, based on 10CFR50.67, for the analysis of the fuel handling accident. The new analysis follows the guidance of Regulatory Guide 1.183 (Reference 4). The proposed changes to the technical specifications are supported by the results of the new analysis.

The proposed changes have been evaluated in accordance with 10 CFR 50.91 (a)(1) using the criteria of 10 CFR 50.92 (c) and ENO has determined that this proposed change involves no significant hazards considerations (Attachment I). The proposed changes to the Technical Specification and Bases pages are provided in Attachment II and a description of the analysis is provided in Attachment III.

Adoption of TSTF-68 and TSTF-312 involves establishing administrative controls regarding the ability to close airlock doors and penetration flow paths, respectively, in the event of a fuel handling accident in containment. Administrative controls are also being proposed to support the requested change for the equipment hatch opening. Attachment IV identifies the commitments being made by ENO, including establishing the administrative controls. ENO requests approval of the proposed amendment by January 17, 2003 to support use of the new requirements during Refueling Outage 12, scheduled to begin in March 2003. Once approved, the amendment will be implemented within 60 days. If you have any questions or require additional information, please contact Mr. Kevin Kingsley, NRR Project Manager at 914-734-6034.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 6-5-2002

Very truly yours,

Robert J. Barrett

Robert J. Barrett
Vice President, Operations – IP3
Indian Point 3 Nuclear Power Plant

Attachments:
I. Analysis of Proposed Technical Specification Changes
II. Proposed Technical Specification and Bases Changes (markup)
III. Westinghouse Analysis of a Fuel Handling Accident at Indian Point 3
IV. Commitments for Adoption of Proposed Technical Specification Changes

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ATTACHMENT I TO IPN-02-044

ANALYSIS OF PROPOSED TECHNICAL SPECIFICATION CHANGE REGARDING SELECTIVE ADOPTION OF ALTERNATE SOURCE TERM FOR FUEL HANDLING ACCIDENTS

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-64, Docket No. 50-286 for Indian Point Nuclear Generating Unit No. 3.

The proposed changes to Section 3.9.3 of the Indian Point 3 Technical Specifications and the proposed relocation of Sections 3.3.8 and 3.7.13 are based on the adoption of an alternate source term for the analysis of a fuel handling accident in accordance with 10CFR50.67. The new analysis demonstrates that applicable dose criteria are met with no credit for the isolation capabilities of the containment building or the fuel storage building. Therefore, certain changes to the plant technical specifications can be made to reflect the results of the analysis. In general, the proposed changes allow containment closure requirements to be relaxed under administrative control, during the movement of irradiated fuel in containment. Also, the requirements of the fuel storage building emergency ventilation system (FSBEVS) and associated actuation instrumentation are being relocated from the technical specifications to a licensee-controlled document. Several of the proposed changes adopt NRC approved Technical Specification Task Force (TSTF) revisions to the Standard Technical Specifications, NUREG-1431. The other changes, for which relevant TSTFs are not available, are based on similar amendment requests approved by the NRC on a plant specific basis. Entergy Nuclear Operations, Inc (ENO) is also making commitments related to establishing the administrative controls for containment closure and for relocating the FSBEVS requirements.

2.0 PROPOSED CHANGE

a. Indian Point 3 Technical Specification LCO 3.9.3.a currently states:

   "The equipment hatch closed and held in place by at least four bolts or the equipment hatch opening is closed using an equipment hatch closure plate that may include a closed personnel access door;"

   LCO 3.9.3.a is revised to state:

   "The equipment hatch opening is capable of being closed;"

b. Indian Point 3 Technical Specification LCO 3.9.3.b currently states:

   "One door in each air lock closed;"

   LCO 3.9.3.b is revised to state:

   "One door in each air lock is capable of being closed;"

c. The following note is being added to Indian Point 3 Technical Specification LCO 3.9.3.c:

   "Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls."

d. Indian Point 3 Technical Specification LCO 3.9.3.d and 3.9.3.e and the associated note are being deleted in entirety. This proposed change also deletes the associated surveillances (SR 3.9.3.2 and 3.9.3.4).

e. Indian Point 3 Technical Specification LCO 3.9.3, Applicability currently states:

   "During CORE ALTERATIONS,
   During movement of irradiated fuel assemblies within containment."

   Applicability is being revised to state:

   "During movement of irradiated fuel assemblies within containment."

   This proposed change also deletes Required Action A.1.

f. Indian Point 3 Technical Specification LCO 3.7.13 defines operability requirements for the Fuel Storage Building Emergency Ventilation System (FSBEVS) and LCO 3.3.8 defines operability requirements for the actuation instrumentation for the FSBEVS. These requirements are being removed from the Technical Specifications and relocated to a licensee-controlled document (Technical Requirements Manual).

Proposed revisions to the Bases related to the above changes are included in Attachment II, for information.

3.0 BACKGROUND

The Indian Point 3 licensing basis for the fuel handling accident (FHA) radiological consequences analyses for Chapter 14 of the FSAR (Reference 1) is currently based on the methodologies and assumptions that are derived from TID-14844 (Reference 2). 10 CFR 50.67 allows licenses to revise the current source term used in radiological analyses. Regulatory Guide 1.183 (Reference 3) provides methods and assumptions that may be used in adopting an alternate source term for use in evaluating the radiological consequences of various hypothetical accident scenarios, including the FHA. This amendment request proposes to adopt an alternate source in accordance with 10 CFR 50.67, using the guidance of Regulatory Guide 1.183. The scope of this change in using the alternate source term is limited to the FHA in containment and the fuel storage building. Use of the alternate source term to revise the radiological analyses for other design basis accidents may be submitted separately. Affected sections of the FSAR will be revised in accordance with 10CFR50.71 to reflect the new analysis assumptions, methods, and results as compared to the regulatory acceptance criteria. In addition, this amendment request proposes changes to the Indian Point 3 Technical Specifications to implement the new accident analysis results. These proposed changes include the adoption of TSTFs 51, 68, and 312 (References 4, 5, and 6). The proposed changes provide a means to improve the efficiency of certain activities performed during a refueling outage, with no adverse affect on plant personnel or public safety. The primary result of the revised analysis, based on the alternate source term, is a demonstration that regulatory dose limits are satisfied with no credit taken for the retention of fission products by the containment building or the fuel storage building or the ventilation / filtration systems for those buildings.

Implementation of the alternate source term methodology for analysis of the FHA in containment and the fuel storage building supports the following proposed changes to the technical specifications:

a. <u>Equipment hatch opening capable of being closed</u>

Technical Specification 3.9.3.a currently requires the equipment hatch or an equipment hatch closure plate to be installed. The proposed change will allow the equipment hatch opening to be open if it is capable of being closed. This allowance will provide additional flexibility in scheduling and performing outage activities with less impact on the critical path duration. Activities that depend on service lines (e.g., electricity, water, air) fed from outside containment and the transport of materials into and out of containment can proceed safely in parallel with fuel movement.

The analysis of the fuel handing accident using the AST methodology demonstrates that regulatory dose limits are satisfied with no credit for protection provided by the containment building. Therefore, allowing the equipment hatch opening to be open is consistent with safety analysis assumptions. Technical specification requirements regarding the equipment hatch opening will be retained and administrative controls will be established to ensure closure of the equipment hatch opening in the event of a FHA to minimize potential migration of fission product activity to the outside atmosphere. The equipment hatch opening can be closed by the normal Equipment Hatch that is used when the plant is at power operation, or it can be closed with the use of the Outage Equipment Hatch (OEH) that has been designed and evaluated for use with the plant in Mode 6, Refueling. The OEH also has a personnel access door and penetrations for service lines such as power cords and air / water hoses. The proposed new specification will allow the Equipment Hatch, the OEH, or the access doors / penetrations in these hatches to be open and capable of being closed during movement of irradiated fuel. Although the safety analysis does not credit any time limit for closure, good practice dictates that this should be accomplished with minimal effort and under the authority of an individual designated to direct the response to a FHA. The administrative controls proposed by ENO to satisfy the 'capable of being closed' requirement will ensure that the equipment hatch opening can be closed within 30 minutes from the time that the designated individual directs that this action be taken. Administrative controls require that any obstructions (e.g., hoses) placed in the opening(s) can be readily removed and that specified individuals are identified and available on site to close these opening(s) when directed. This application for amendment includes a commitment from ENO to establish the administrative controls. Implementation of administrative controls is reflected in the proposed changes to the Bases. In addition, the Bases for the surveillance requirement associated with this LCO are expanded to include verification that if the openings are not closed, that they are capable of being closed.

b. Air lock doors capable of being closed

Technical Specification 3.9.3.b currently requires one door in each airlock to be closed. The proposed change adopts TSTF-68, Rev 2, which has been approved by the NRC and is incorporated into Revision 2 of NUREG-1431. This change provides the option to allow both doors in any containment personnel airlock to remain open during the movement of irradiated fuel assemblies. As stated in the 'Reviewers Note' established by this TSTF, adopting this allowance is based on dose calculations that indicate acceptable radiological consequences and commitments from the licensee to implement administrative controls regarding prompt closure of a door in each airlock in the event of a FHA. The consequences of a FHA have been evaluated using the alternate source term methodology and acceptable radiological consequences have been demonstrated by analysis. The analysis assumptions bound the condition of having the airlock doors open, by taking no credit for holdup of fission products by the containment building. This request for amendment to the Indian Point 3 Technical Specifications includes a commitment by Entergy Nuclear Operations, Inc to implement the required administrative controls. Implementation of administrative controls is reflected in the proposed changes to the Bases.

c. Penetrations open under administrative control

Technical Specification 3.9.3.c currently requires containment penetrations to be closed, except that an operable containment purge system isolation valve can be open. The proposed change adopts TSTF-312, Rev 1, which has been approved by the NRC and is incorporated into Revision 2 of NUREG-1431. This change adds a Note to the LCO allowing penetration flow paths to be unisolated under administrative control. The adoption of this allowance is contingent on meeting the terms of a 'Reviewers Note' as discussed in item b. The revised dose analysis for the FHA bounds the condition of having containment penetrations unisolated by taking no credit for holdup of fission products by the containment building. As stated in item b, ENO is committing to administrative controls as implemented by proposed changes to the Bases.

d. Delete ventilation requirements at less then 550 hours subcritical

Amendment 175 to the Indian Point 3 Technical Specifications was issued on July 15, 1997 to allow for the use of Vantage+ fuel beginning with fuel cycle 10. NRC review of that amendment included independent dose analysis of the FHA to account for the larger design radial peaking factor that would be used with the new fuel type. On the basis of that analysis, the NRC staff determined that HEPA and charcoal filtration must be used during movement of Vantage+ fuel for the first 550 hours following reactor shutdown. This analysis used the existing licensing basis analysis methodology.

The new analysis, that adopts the alternate source term methodology and acceptance criteria, demonstrates acceptable dose consequences with no credit for HEPA and charcoal filtration of the containment ventilation system. Therefore, the requirements of LCO 3.9.3.d and 3.9.3.e regarding the containment purge and containment pressure relief lines with the reactor subcritical for less than 550 hours, can be deleted. This proposed change also involves the deletion of the surveillance requirements (SR 3.9.3.2 and 3.9.3.4) that are associated with verifying the affected portion of the LCO. As with the changes proposed in items a, b, and c, this change would be subject to a confirmatory dose calculation by the NRC staff, consistent with the Reviewer's Note in section 3.9.4 of the Standard Technical Specification Bases. However, there would be no administrative controls related to implementing this change.

e. Modify applicability to remove 'during core alterations'

The applicability for Technical Specification 3.9.3 currently includes 'During CORE ALTERATIONS' and 'During movement of irradiated fuel assemblies within containment'. The proposed change adopts a portion of TSTF-51, Rev 2, which has been approved by the NRC and is incorporated into Revision 2 of NUREG-1431. This change adopts the deletion of 'During CORE ALTERATIONS' from the applicability of 3.9.3. The TSTF supports the deletion of this applicability from various other specifications and further modifies the affected applicability by defining 'recently' irradiated fuel. Although these changes are justified at Indian Point 3 based on the new analysis results, the full scope of TSTF-51 is not being adopted at this time. ENO may propose to adopt the balance of TSTF-51 in a future license amendment request. Accidents postulated to occur during core alterations may include inadvertent criticality (due to control rod removal error or continuous control rod withdrawal error during refueling or boron dilution) and the inadvertent loading of, and subsequent operation with, a fuel assembly in an improper location. These events are not postulated to result in fuel cladding integrity damage. Therefore the requirements of Technical Specification 3.9.3 that are in place to mitigate the consequences of a FHA, which is postulated to result in fuel cladding damage, are not required during activities involving only core alterations.

f. Relocate requirements for fuel storage building emergency ventilation system

Technical Specification 3.7.13 currently requires the fuel storage building emergency ventilation system (FSBEVS) to be operable when moving irradiated fuel assemblies in the FSB. The current safety function for this system is to limit the dose consequences of a FHA in the FSB to the acceptance criteria of 10CFR100. Therefore, this system is currently in the Technical Specifications because it satisfies Criterion 3 of 10CFR50.36.

The analysis of a FHA using the alternate source term methodology bounds a dropped fuel assembly in containment and the FSB. The analysis demonstrates that regulatory dose limits applicable to the AST are met with no credit for holdup by the FSB structure or filtration by the FSBEVS. Therefore, the FSBEVS no longer meets 10CFR50.36 criteria and can be relocated from the technical specifications to a licensee controlled document. In addition, Technical Specification 3.3.8 requires the actuation instrumentation for the FSBEVS to be operable when moving irradiated fuel assemblies in the FSB. These requirements can be relocated to the TRM for the same reason that Technical Specification 3.7.13 is being relocated. ENO is committing to relocate the existing FSBEVS and actuation instrumentation requirements to the Technical Requirements Manual as part of implementing this proposed change.

4.0 TECHNICAL ANALYSIS

The amendments identified in this document are based on the results of a revised Fuel Handling Accident (FHA) analysis (Attachment III) performed by Westinghouse Electric Corporation. The analysis has been reviewed and accepted by ENO to replace the existing licensing basis FHA analysis for Indian Pont 3. This analysis characterizes the dose resulting from a dropped fuel assembly, either in the Vapor Containment Building (CB) or the Fuel Storage Building (FSB). The analysis takes no credit for isolation or filtration in either location. Accordingly, the proposed amendment modifies requirements for containment closure during fuel handling, provided that all penetrations are capable of prompt closure as defined in the proposed changes to the Bases.

Technical Specification 3.9.3 separately lists three containment penetrations; equipment hatch opening, airlocks, and other penetrations. Of these, the most significant is the equipment hatch opening, since it communicates directly with the outside environment. In the event of a FHA, the equipment hatch closure would commence immediately under the supervision of the Manager in charge of refueling, who is stationed within the CB and would know of the FHA immediately. All piping and cabling passing through the hatch opening will be configured to support easy removal (i.e., quick disconnects or other rapid disassembly methods) so that the opening can be promptly closed. Once the equipment hatch opening is closed, the primary flow path between the CB and the outside atmosphere has been secured. Any other open penetrations that communicate directly to the atmosphere will also be placed on first priority for prompt closure. The personnel airlocks can be easily closed in a matter of minutes, once the VC has been evacuated of non-essential personnel. The remaining penetrations do not communicate directly with the outside environment and are therefore less likely to contribute to the migration of fission product activity to the outside atmosphere. To summarize, the administrative controls ensure that all open penetrations can be securely closed in a safe and orderly manner, without undue personnel hazard or safety risk.

The Westinghouse analysis takes no credit for automatic isolation of the Control Room (CR) HVAC system. The analysis assumes normal CR ventilation is in effect until the high radiation alarm actuates (the analysis conservatively assumes one minute for this to occur). After that, an interval of twenty minutes is assumed for operator action to manually establish HVAC emergency mode. This is a conservative assumption, since the CR operators would have been made aware of the FHA immediately via normal refueling communications channels that are required by the FSAR and plant procedures. Once emergency HVAC is established, the analysis assumes unfiltered inleakage over a range of 1000 to 1800 cfm.

The Westinghouse analysis provides a series of post-criticality times for comparative evaluation of resultant doses. The earliest analyzed FHA time is 48 hours subcritical, and the latest is 84 hours. The 48-hour interval results in doses within the acceptance criteria assuming 1000 cfm unfiltered inleakage, and the 84-hour interval supports 1800 cfm unfiltered inleakage. ENO intends to implement the latter time period (84 hours) until such time as the CR inleakage rate can be quantified via tracer gas test. Due to the relatively small size of the CR at Indian Point Unit 3, it is anticipated that measured inleakage will be well below 1800 cfm, resulting in lower calculated control room doses.

ENO has implemented the test standard (ASTM D3803-1989) of Generic Letter 99-02 for verification of the efficiency of the installed charcoal filters. This test method is enforced by the

FSAR pending completion of a proposed amendment to the Ventilation Filter Testing Program requirements in the Technical Specifications.

The analysis is performed in accordance with Regulatory Guide (RG) 1.183 and uses the approved Alternate Source Term (AST) methodology. It has been performed in a manner similar to that of other Westinghouse plants, such as Kewaunee and Shearon Harris. The analytical methodology addresses the issues identified in NRC Regulatory Issue Summary Letter 2001-19.

Other assumptions used in the analysis include:

1. The iodine species in the SFP are 99.85% elemental and 0.15% organic. This is based on splitting the activity leaving the damaged fuel into 95% cesium iodide, 4.85% elemental iodine and 0.15% organic iodine.
2. Based on the Technical Specifications requirement for a depth of at least 23 feet over the fuel, the decontamination factor is assumed to be 200.
3. The cesium and rubidium released from the damaged fuel rods are assumed to remain in a nonvolatile form and would not be released from the pool.
4. The radionuclides that are contributors to the dose analysis are isotopes of xenon, krypton, and iodine. A list of the specific isotopes evaluated can be found in Table 6 of Attachment III.
5. The scenario assumes that one fuel assembly is dropped and that every rod within that assembly is damaged to the extent that all gap activity is released. It should be noted that this assumption is the traditional method of analysis but remains exceedingly conservative, in that relatively few, if any, rods are expected to fail subsequent to a FHA (as noted in FSAR Section 14.2-1, Page 14.2-4).
6. Meteorological dispersion uses the existing licensing basis method using the Sutton dispersion model as described in the FSAR.

In accordance with RG 1.183, gap fractions have been applied for the damaged fuel assembly, which is assumed in the analysis to be the highest powered assembly in the reactor core. RG 1.183 supplies gap fractions suitable for use in AST analysis, provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft together with having a burnup of greater than 54 GWD/MTU. Due to the nature of the IP3 two-year fuel cycle design, ENO estimates that up to 10% of the rods in the peak assembly may exceed this combined heatrate/burnup criteria. The analysis assumes that 25% of the rods exceed the heatrate/burnup criteria and applies a higher gap fraction inventory consistent with RG 1.25 (as modified by the direction of NUREG/CR-5009) is used for these rods.

In order to ensure that future cores will be valid under the AST analysis, a design criterion will be added to the Reload Safety Analysis Checklist (RSAC). This will require confirmation that no fuel assembly contains more than 25% of fuel rods with heat rate greater than 6.3 kw/ft and a burnup exceeding 54 GWD/MTU. Incorporation of this criterion into the RSAC makes it a design constraint to be applied during every subsequent core reload.

It should be noted that the actual time to offload the core will also be influenced by the thermal load on the Spent Fuel Pit (SFP) cooling system. Limitations for the SFP are currently provided in Section 9.3.3 of the FSAR, which define the design capability of the present cooling system to be 35 MBTU/hr. Core offload will be controlled to ensure that this limiting heat load is not exceeded. This will be implemented by cycle specific analysis and administrative controls.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change involves the reanalysis of a fuel handing accident (FHA) in containment and in the fuel storage building. The new analysis, based on the Alternate Source Term (AST) in accordance with 10 CFR 50.67, will replace the existing analysis based on methodologies and acceptance criteria in place when Indian Point 3 was originally licensed. As a result of the new analysis, changes to the Technical Specifications are proposed which take credit for the new analysis results.

The proposed changes to the technical specifications modify requirements regarding containment closure during movement of irradiated fuel assemblies in containment and relocate requirements for the fuel storage building emergency ventilation system from the technical specifications to a licensee controlled document. The proposed changes do not involve physical modifications to plant equipment and do not change the operational methods or procedures used for moving irradiated fuel assemblies. As such, there are no accident initiators affected by the proposed amendment. The revised requirements apply only when the plant is in a refueling condition (Mode 6), and specifically only when irradiated fuel is being moved. Previously evaluated accidents with the plant in other conditions ranging from cold shutdown (Mode 5) through power operation (Mode 1) are not affected. The AST methodology is used to evaluate a FHA that is postulated to occur during fuel movement activities in the containment building and the fuel storage building. The analysis follows the guidance of the NRC Regulatory Guide 1.183 and uses the acceptance criteria of the NRC Standard Review Plan (NUREG 0800) for offsite doses and General Design Criteria 19 for control room personnel. The analysis demonstrates that the dose consequences meet regulatory acceptance criteria. The accident analysis conservatively assumes that the containment building and the fuel storage building, including ventilation filtration systems for those building does not diminish or delay the assumed fission product release. The analysis does take credit for, and technical specifications enforce, the presence of 23 feet of water over the irradiated fuel while fuel movement activities are being performed. The analysis also takes credit for, and the technical specification bases enforce a fuel decay time of at least 84 hours. In addition, administrative controls are put in place to provide for closure of containment openings in the event of a FHA. Use of an alternate analysis method does not affect fuel parameters or the equipment used to handle the fuel. The proposed changes to the technical specifications reflect assumptions made in the analysis.

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

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

Response: No.

The proposed amendment involves the use of an alternate analysis methodology for the evaluation of the dose consequences from a FHA that is postulated to occur in either the containment building or the fuel storage building (FSB). The analysis demonstrates that containment closure conditions and operation of the containment purge filtration system are not required to maintain dose consequence within regulatory limits following a postulated FHA in containment. Therefore the new analysis supports proposed changes to requirements for containment closure during movement of irradiated fuel assemblies in containment. The analysis results also demonstrate that operation of the fuel storage building emergency ventilation system is not required to maintain dose consequences within regulatory limits following a postulated FHA in the FSB. The containment closure components (e.g., equipment hatch, personnel airlock doors, and various containment penetrations) and filtration systems are not accident initiators. The proposed changes do not involve the addition of new systems or components nor do they involve the modification of existing plant systems. The proposed changes do not affect the way in which a FHA is postulated to occur.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The existing dose analysis methodology and assumptions demonstrates that the dose consequences of a FHA are within regulatory limits for whole body and thyroid doses as established in 10 CFR 100. The alternate dose analysis methodology and assumptions also demonstrates that the dose consequences of a FHA are within regulatory limits. The limits applicable to the alternate analysis are established in 10 CFR 50.67 in conjunction with the TEDE (total effective dose equivalent) acceptance directed in Regulatory Guide 1.183. The acceptance criteria for both dose analysis methods have been developed for the purpose of evaluating design basis accidents to demonstrate adequate protection of public health and safety. An acceptable margin of safety is inherent in both types of acceptance criteria.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy Nuclear Operations, Inc. concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements / Criteria

The proposed changes have been evaluated to determine compliance with applicable regulatory requirements.

The revised analysis for the fuel handling accident is based on 10 CFR 50.67 and uses the regulatory guidance of Regulatory Guide 1.183 and Standard Review Plan (NUREG 0800) Section 15.0.1. The analysis demonstrates compliance with these regulatory requirements and criteria. Use of the new analysis method replaces 10CFR100 as the applicable dose acceptance criteria for the fuel handling accident. Based on the approach of selective adoption of the alternate source term, the criteria of 10CFR100 continue to be applicable and are satisfied for other design basis accidents.

GDC 19 requires that holders of an operating license using an alternative source term under 10 CFR 50.67 shall meet the requirements of this criterion by ensuring that radiation exposures to control room occupants shall not exceed 5 rem TEDE. The analysis provided to support the requested Technical Specification changes demonstrates that this criterion is satisfied.

ENO has determined that the proposed changes do not require any exemptions or relief from regulatory requirements other then the change requested to Technical Specification Sections 3.7.13 and 3.9.3. The proposed use of an alternate source term to evaluate the radiological consequences of a fuel handling accident results in a change to the existing licensing basis analysis described in the FSAR. In accordance with 10CFR50.71, ENO will update the FSAR to reflect the proposed new analysis method. The changes to the technical specifications incorporate assumptions used in the new analysis. The analysis results demonstrate that the affected technical specifications are no longer needed to satisfy Criterion 3 of 10 CFR 50.36. Therefore Technical Specification 3.7.13 is being relocated to a licensee controlled document. ENO proposes to retain Technical Specification 3.9.3, modified to reflect analysis assumptions and using the guidance of the latest version (Revision 2) of the Standard Technical Specifications.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

In 1996, the NRC issued SECY-96-242 to outline an approach to allow the use of a revised accident source term at operating reactors. Subsequently, several operating plants were identified to participate in a pilot program to submit license amendments to modify various aspects of their licensing basis, using the alternate source term. In 1998, the NRC approved an application for the Perry Nuclear Plant. Since that time other plants have submitted and obtained approval of similar applications. In addition, certain of the proposed changes to the Indian Point 3 Technical Specifications are based on NRC-approved generic changes (TSTF) to the Standard Technical Specifications (NUREG 1431) that are applicable to Indian Point 3. The following table identifies recent examples of approved changes that are similar to those proposed in this application request.

ITEM	SUMMARY	RECENT APPROVAL
a	Equipment hatch capable of being closed	Example amendment that uses the alternate source term methodology to support this change is Ft. Calhoun Amendment 204 issued March 2002.
b	Personnel airlock doors capable of being closed	This is in accordance with TSTF 68. Example adoption is Watts Bar Amendment 26 issued August 2000.
c	Containment flow paths open under administrative control.	This is in accordance with TSTF 312. Example adoption is Shearon Harris Amendment 104 issued July 2001.
d	Delete containment ventilation filtration requirements with reactor subcritical less than 550 hours.	This requirement was added to the Indian Point 3 Technical Specifications as a plant specific condition for use of a new fuel type. The alternate source term analysis supports deleting this requirement, which is consistent with the Standard Technical Specifications.
e	Delete 'during Core Alterations' from applicability.	This is in accordance with TSTF 51. Example adoption is Watts Bar amendment 35 issued January 2002.
f	Relocate Fuel Storage Building Emergency Ventilation System and associated actuation instrumentation requirements out of Technical Specifications.	Example amendment that uses the alternate source term methodology to support this change is Surry amendment 230 issued March 2002.

7.0 REFERENCES

1. Indian Point 3 Final Safety Analysis Report, Section 14.2.1, "Fuel Handling Accidents."

2. J. J. DiNunno, et al., "Calculation of Distance Factors for Power and Test Reactor Sites," USAEC TID-14844, U. S. Atomic Energy Commission (now USNRC), 1962.

3. Regulatory Guide 1.183; "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," dated July 1, 2000.

4. TSFT-51, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations," NRC approved November 1, 1999.

5. TSTF-68, Revision 2, "Containment Personnel Airlock Doors Open During Fuel Movement," NRC approved August 16, 1999.

6. TSTF-312, Revision 1, "Administratively Control Containment Penetrations," NRC approved August 16, 1999.

ATTACHMENT II TO IPN-02-044

MARKUP OF TECHNICAL SPECIFICATION AND BASES

FOR PROPOSED CHANGES REGARDING

SELECTIVE ADOPTION OF ALTERNATE SOURCE TERM

FOR FUEL HANDLING ACCIDENTS

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

opening is capable of being closed;

a. The equipment hatch closed and held in place by at least four bolts or the equipment hatch opening is closed using an equipment hatch closure plate that may include a closed personnel access door;

DELETE

is capable of being

b. One door in each air lock closed;

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

1. closed by a manual or automatic isolation valve, a blind flange, or equivalent, or

Insert NOTE, See next page

2. capable of being closed by OPERABLE Containment Purge Isolation System.

DELETE

---NOTE---
LCO 3.9.3.d and LCO 3.9.3.e are not required to be met if the reactor has been subcritical for ≥ 550 hours.

---

d. The Containment Purge System flow path shall be either:

1. closed by a manual or automatic isolation valve, a blind flange, or equivalent, or

2. aligned to discharge through the HEPA filters and charcoal adsorbers.

e. The Containment Pressure Relief Line shall be closed by a manual or automatic isolation valve, a blind flange, or equivalent.

DELETE

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.~~2~~ 1	Suspend movement of irradiated fuel assemblies within containment.	Immediately

DELETE

Insert for page 3.9.6 – 1:

-----------------------------NOTE-----------------------------
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	---------------------- NOTE ---------------------- Not required to be met if the reactor has been subcritical for ≥ 550 hours. --- Verify Containment Purge System is either: a. closed by a manual or automatic isolation valve, blind flange, or equivalent, or b. aligned to discharge through the HEPA filters and charcoal absorbers.	7 days
SR 3.9.3.~~3~~ 2	Verify each required containment purge system valve actuates to the isolation position on an actual or simulated actuation signal.	92 days
SR 3.9.3.4	---------------------- NOTE ---------------------- Not required to be met if the reactor has been subcritical for ≥ 550 hours. --- Perform required Containment Purge System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

DELETE

DELETE

REMOVE FROM TECHNCIAL SPECIFICATIONS
RELOCATE TO TECHNICAL REQUIREMENTS MANUAL

3.3 INSTRUMENTATION

3.3.8 Fuel Storage Building Emergency Ventilation System (FSBEVS) Actuation Instrumentation

LCO 3.3.8 FSBEVS manual and automatic actuation instrumentation shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel in the fuel storage building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Manual or automatic FSBEVS actuation instrumentation inoperable.	A.1 Place FSBEVS in operation.	Immediately
	OR	
	A.2 Suspend movement of irradiated fuel in the fuel storage building.	Immediately

REMOVE FROM TECHNCIAL SPECIFICATIONS
RELOCATE TO TECHNICAL REQUIREMENTS MANUAL

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.8.1 Perform CHANNEL CHECK.	24 hours
SR 3.3.8.2 Perform COT.	92 days
SR 3.3.8.3 Perform CHANNEL CALIBRATION.	24 months

REMOVE FROM TECHNCIAL SPECIFICATIONS
RELOCATE TO TECHNICAL REQUIREMENTS MANUAL

3.7 PLANT SYSTEMS

3.7.13 Fuel Storage Building Emergency Ventilation System (FSBEVS)

LCO 3.7.13 FSBEVS shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel storage building.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. FSBEVS inoperable.	A.1 Suspend movement of irradiated fuel assemblies in the fuel storage building.	Immediately

REMOVE FROM TECHNCIAL SPECIFICATIONS
RELOCATE TO TECHNICAL REQUIREMENTS MANUAL

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify FSBEVS charcoal filter bypass dampers are installed.	92 days
SR 3.7.13.2	Operate FSBEVS for ≥ 15 minutes.	31 days
SR 3.7.13.3	Perform required FSBEVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.4	Verify FSBEVS actuates on an actual or simulated actuation signal.	92 days
SR 3.7.13.5	Verify FSBEVS can maintain a pressure ≤-0.125 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate ≤ 20,000 cfm.	24 months

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed, except for the OPERABLE Purge System Penetration. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

Insert 1-A

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

In lieu of maintaining the equipment hatch in place for containment closure, a temporary closure device may be used to maintain containment closure during core alterations or during

DELETE

(continued)

BASES

BACKGROUND (continued)

DELETE

movement of irradiated fuel assemblies within containment. The temporary closure device may provide penetrations for temporary services or personnel access. The temporary closure device will be designed to withstand a seismic event and designed to withstand a pressure which ensures containment closure during refueling operations. The closure device will provide the same level of protection as that of the equipment hatch for the fuel handling accident by restricting direct air flow from the containment to the environment.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, ~~containment closure is required; therefore,~~ the door interlock mechanism may remain disabled, but one air lock door must always remain capable of being closed.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge System consists of the 36-inch containment purge supply and exhaust ducts. The supply system includes roughing filters, heating coils, fan and a containment penetration with two butterfly valves for isolation. The exhaust system includes a containment penetration with two butterfly valves for isolation and can be aligned to discharge to the

(continued)

BASES

BACKGROUND (continued)

atmosphere through the plant vent either directly or through the Containment Purge Filter System (i.e., a filter bank with roughing, HEPA and charcoal filters).

The Containment Purge System must be isolated when in Modes 1, 2, 3 or 4 in accordance with requirements established in LCO 3.6.3, Containment Isolation Valves. In Modes 5 and 6, the Containment Purge System may be used for containment ventilation. When open, the Containment Purge System isolation valves are capable of closing in response to the detection of high radiation levels in accordance with requirements established in LCO 3.3.6, (Ref 1). Containment Purge and Pressure Relief Isolation Instrumentation Despite this isolation capability, the Containment Purge System must be aligned to discharge through the Containment Purge Filter System during CORE ALTERATIONS or movement of irradiated fuel until the reactor has been shutdown for a specified minimum number of hours.

DELETE

The Containment Pressure Relief Line (i.e., Containment Vent) consists of a single 10-inch containment vent line that is used to handle normal pressure changes in the Containment when in Modes 1, 2, 3 and 4 (Ref. 1). The Containment Pressure Relief Line is equipped with three quick-closing butterfly type isolation valves, one inside and two outside the containment which isolate automatically in accordance with requirements established in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation", and LCO 3.3.6, "Containment Purge System and Pressure Relief Line Isolation Instrumentation." Although the Containment Pressure Relief Line discharges to the atmosphere via the Containment Auxiliary Charcoal Filter System (i.e., a filter bank with roughing, HEPA and charcoal filters), the Containment Pressure Relief Line must remain isolated during CORE ALTERATIONS or movement of irradiated fuel until the reactor has been shut down for a specified minimum number of hours. The Containment Pressure Relief Line must remain isolated because the Containment Auxiliary Charcoal Filter System is not required to be tested in accordance with Specification 5.5.10, Ventilation Filter Test Program.

(continued)

or may be unisolated under administrative control

BASES

BACKGROUND (continued)

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved ~~in accordance with 10 CFR 50.59~~ and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements.

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The release of radioactivity from the containment following a fuel handling accident is limited by the following:

a) The requirements of LCO 3.9.6, "Refueling Cavity Water Level;"

and,

moving irradiated fuel.

b) The minimum decay time of ~~145~~ 84 hours prior to ~~CORE ALTERATIONS; and,~~

Insert 4-A

DELETE

c) The requirements of this LCO to either isolate the Containment Purge System or align the system to discharge through the HEPA filters and charcoal adsorbers for a minimum of first 550 hours following the reactor shutdown.

This combination of requirements ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

DELETE

The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36.

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge system penetrations. For the OPERABLE containment purge system penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge isolation instrumentation. Additionally, the requirement to isolate the Containment Purge System or align the system to discharge through the HEPA filters and charcoal absorbers for a minimum of the first 550 hours following the reactor shutdown is required to limit offsite radiation exposure to within required limits. The Containment Pressure Relief Line must remain isolated because the Containment Auxiliary Charcoal Filter System is not required to be tested in accordance with Specification 5.5.10, Ventilation Filter Test Program. The OPERABILITY requirements for this LCO ensure that the automatic purge system valves meet the assumptions used in the safety analysis to ensure that releases through the valves are filtered and can be terminated, such that radiological doses are within the acceptance limit.

or capable of being closed

DELETE

Insert 5-A

APPLICABILITY

The containment penetration requirements are applicable during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment are not being conducted, the

(continued)

BASES

APPLICABILITY (continued)

potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 ~~and A.2~~

If the containment equipment hatch opening, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge system isolation instrumentation not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations is either closed or capable of being closed under administrative control ~~required to be in its closed position is in that position~~. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Insert 7-A

SR 3.9.3.1 (continued)

verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

DELETE

SR 3.9.3.2

This SR requires periodic verification every 7 days that the Containment Building Purge System is either isolated or aligned to discharge through the HEPA filters and charcoal adsorbers. This SR is needed because it requires periodic verification that LCO 3.9.3.d is being met. A Note provides the allowance that this SR is not required to be performed or met if the reactor has been subcritical for ≥ 550 hours. These restrictions ensure that the offsite dose limit for a fuel handling accident of 75 rem to the thyroid at the exclusion area boundary (i.e., 25 percent of the 10 CFR Part 100 limit of 300 rem) is met by either filtering any release from the containment or by allowing a greater decay time before fuel handling activities are permitted.

DELETE

SR 3.9.3.~~3~~ 2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on an actual or simulated high radiation signal. The 92 day Frequency ensures that this SR is performed prior to this function being required and periodically thereafter. In LCO 3.3.6, the Containment Purge system isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months a CHANNEL CALIBRATION is performed. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.~~3~~ 2 (continued)

These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

DELETE

SR 3.9.3.4

This SR verifies that the required Containment Building Purge System testing is performed in accordance with Specification 5.5.10, Ventilation Filter Test Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. FSAR, Section 5.3.

2. FSAR, Section 14.2.

~~3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.~~

add new references:

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

4. 10 CFR 50 Appendix A, "General Design Criteria," Criterion 19, Control Room.

INSERTS – BASES PAGES

Insert 1-A (page B 3.9.3 - 1)

In MODE 6, the containment serves to contain fission product radioactivity that may be released following a fuel handling accident, although this feature is not required to meet regulatory dose limits. Containment penetrations consist of the equipment hatch opening, personnel airlocks, and penetration flow paths, including the containment purge system. The requirements of LCO 3.9.3 and related administrative controls, reduce the potential for migration of fission product radioactivity out of containment following a fuel handling accident.

The equipment hatch opening provides a means for moving large equipment and components into and out of containment. The main Equipment Hatch is part of the containment pressure boundary used to meet containment integrity requirements in MODES 1, 2, 3, and 4. The main Equipment Hatch may also be used for closure of the equipment hatch opening during movement of irradiated fuel assemblies. Good engineering practice dictates that the main Equipment Hatch be held in place by at least four equally-spaced bolts. In lieu of the main Equipment Hatch, a temporary closure device (e.g., Outage Equipment Hatch, OEH) may be used for closure of the equipment hatch opening during movement of irradiated fuel assemblies. The OEH will provide the same level of protection as the main Equipment Hatch for the fuel handling accident by restricting direct airflow from the containment to the environment.

Administrative controls provide for prompt closure of the equipment hatch opening in the event of a fuel handling accident. The main Equipment Hatch or the OEH must be installed during movement of irradiated fuel assemblies in order the meet the time limits established by administrative control for prompt closure following a fuel handling accident. The main Equipment Hatch and the OEH may be equipped with penetrations that can be used for personnel access and/or service lines. The capability to promptly close these openings is maintained under administrative control.

Insert 4-A (page B 3.9.3 - 4)

These requirements ensure that applicable dose limits, based on use of the alternate source term methodology per 10 CFR 50.67, are satisfied. Dose limits for the Exclusion Area Boundary and the Low Population Zone are established by Regulatory Guide 1.183 (Reference 3). Dose limits for control room occupants are established by GDC 19 (Reference 4). The analysis of the fuel handling accident does not take credit for retention of fission product radioactivity by the containment building or containment ventilation / filtration systems. However, the requirements of LCO 3.9.3 regarding containment penetrations, serve to minimize the potential for the migration of fission product radioactivity to the outside atmosphere. In addition, administrative controls are used to ensure prompt closure of containment penetrations in the event of a fuel handling accident.

Insert 5-A (page B 3.9.3 - 5)

The equipment hatch opening and the containment personnel airlock doors may be open during movement of irradiated fuel in the containment provided that the equipment hatch opening and one door in each airlock is capable of being closed in the event of a fuel handling accident. Administrative controls are established to ensure this closure capability. In addition, the LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls.

In general, the administrative controls applicable to each of the above containment openings ensure that:

1) Appropriate personnel are aware of the open status of the penetration flow path(s) during movement of irradiated fuel assemblies within containment.

2) Specified individuals are designated and readily available to direct and perform isolation of affected flow paths in the event of a fuel handling accident.

3) Any obstructions (e.g., cables and hoses) that would prevent rapid closure of an open flow path can be quickly removed. Any cables or hoses to be disconnected should not be supplying services that support personnel safety (e.g., breathing air)

Closure of all penetration flow paths shall be performed promptly in the event of a fuel handling accident as determined by the onsite individual designated to direct this activity. The designated individual will take into account the need to take appropriate actions to secure the condition, evacuate containment, and ensure personnel safety. Although the safety analysis does not depend on containment closure to meet regulatory dose limits, closure of all penetrations within 2 hours is desirable to minimize migration of fission product radioactivity from the containment atmosphere to adjacent buildings or the outside atmosphere.

Because the equipment hatch opening provides the most direct path from the containment atmosphere to the outside atmosphere administrative controls ensure that this opening can be closed within 30 minutes from the time the designated individual directs that this action be taken. The containment purge and the containment pressure relief lines also provide a direct path to the outside atmosphere. Therefore, if these paths are open under administrative control, methods must be in place to ensure closure within 30 minutes.

Insert 7-A (page B 3.9.3 - 7)

As such, this surveillance ensures that the migration of fission product radioactivity from the containment atmosphere to the outside atmosphere following a postulated fuel handling accident is minimized.

ATTACHMENT III TO IPN-02-044

WESTINGHOUSE ANALYSIS OF

FUEL HANDLING ACCIDENT

AT INDIAN POINT 3

USING THE ALTERNATIVE SOURCE TERM METHODOLOGY

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

Introduction and Background

A fuel handling accident (FHA) dose analysis has been performed for Indian Point Unit 3 (IP3) to determine the earliest time after shutdown that fuel may be moved and remain within the regulatory dose limits for a FHA. Currently, the IP3 technical specifications require a decay period of 145 hours before fuel assemblies can be moved. A reduction in the time between shutdown and fuel movement is desirable for minimizing outage time.

The methodology in Regulatory Guide (RG) 1.183 (Reference 1) was used to evaluate the radiological dose consequences at the exclusion area boundary (EAB), low population zone (LPZ), and in the control room (CR). The reactor core power level of 3086 MWt was used in the FHA analysis based on 102% of the current licensed power of 3025 MWt.

The FHA currently reported in the UFSAR assumes that the activity released from the damaged assembly is released to the outside environment at a uniform rate over a 2-hour period. With filtration and containment isolation modeled, the analysis allows for refueling operations to begin at 145 hours following shutdown. Without credit for filtration or containment isolation, fuel handling is not allowed until 550 hours after shutdown (Reference 6). The analysis reported herein does not take credit for containment isolation nor does it take credit for the charcoal and HEPA filters in the release paths.

This analysis supports fuel movement earlier than the design basis limit of 145 hours which requires IP3 to either isolate the containment purge system or align the system to discharge through the HEPA filters and charcoal absorbers before fuel movement. It also removes the need for a separate defined time to allow fuel movement without these conditions (currently included in the Technical Specifications as a minimum of 550 hours following reactor shutdown).

Accident Doses

The FHA radiological consequences analysis currently in the UFSAR is based on methodologies and assumptions that are derived from Regulatory Guide 1.25 (Reference 2) and NUREG/CR-5009 (Reference 3).

RG 1.183 provides guidance on the application of alternative source terms (AST) in revising the design basis radiological consequence analyses, as allowed by 10CFR50.67. The alternative source term methodology, as established in RG 1.183, was used in the FHA analysis to calculate the offsite and control room radiological consequences to support early fuel movement.

Using RG 1.183 methodology, all calculated offsite and control room doses are determined to be within the RG 1.183 specified fractions of the 10CFR50.67 limits for decay periods of ≥ 84 hours.

Analysis Inputs and Assumptions

In the FHA analysis, a fuel assembly is assumed to be dropped and damaged during refueling. Activity from the damaged fuel assembly is assumed to be released to the water pool. While the FHA could occur either in the containment building or in the auxiliary building, the same model will apply to both since in the revised analysis no credit is taken for isolation of containment on high radiation or for filtration of releases. The analysis assumes that activity from the damaged fuel assembly is released to the environment over a two-hour period.

This analysis utilizes the total effective dose equivalent (TEDE) dose basis as provided in Reference 1. The TEDE dose is equivalent to the committed effective dose equivalent (CEDE) or inhalation dose plus the acute dose or effective dose equivalent (EDE) dose for the duration of exposure to the cloud. The dose conversion factors (DCFs) used in determining the CEDE dose are from Reference 4 and are given in Table 1. The DCFs used in determining the EDE dose are from Reference 5 and are also provided in Table 1.

The TEDE doses at the EAB and LPZ were determined for the analysis release duration of 2 hours. The interval for determining the control room dose extends beyond the time when the releases are terminated. This accounts for the additional dose to the operators in the control room, which continues as long as significant activity is circulating within the control room envelope.

The offsite breathing rates and the offsite atmospheric dispersion factors used in the radiological calculations are provided in Table 2.

Parameters used in the control room dose calculations are provided in Table 3. These parameters include the normal operation flowrate, emergency operation flowrate, control room volume, filter efficiency, atmospheric dispersion factor, and control room operator breathing rate. The atmospheric dispersion factor is used to determine the activity available at the air intake. The inflow (filtered and unfiltered) to the control room and the recirculation flow in the control room are used to calculate the activity introduced to the control room and cleanup of activity in the control room envelope. The unfiltered in-leakage rate for this analysis has been increased to 1800 cfm from the 10 cfm used in the current UFSAR LOCA analysis. The use of 1800 cfm is an arbitrary bounding value, and its use removes concern about potential high unfiltered in-leakage to the control room.

The major assumptions and parameters used in the analysis are itemized in Table 4. The analysis involves dropping a recently discharged (84-hour decay) PWR fuel assembly. All activity released from the water pool is assumed to be released to the environment within two hours.

Source Term

The core source term based on the time after shutdown is provided in Table 5. The decay time prior to fuel movement used in the analysis is 84 hours. As in the existing licensing basis, it is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all of the gap activity is released. The damaged assembly is assumed to have been operating at the highest fuel rod power level.

Consistent with RG 1.183 (Position 1.2 of Appendix B), the radionuclides considered are xenons, kryptons, halogens, cesiums and rubidiums. The xenons, kryptons, and halogens considered are those listed in Table 5. The cesium and rubidium nuclides are not included because they are non-volatile and are not assumed to be released from the pool.

Table 3 of RG 1.183 identifies gap fractions that are to be used for the analysis of non-LOCA events and states that the specified gap fractions are applicable "provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU". These gap fractions are listed in Table 4.

It is projected that for the end of IP3 Cycle 12, as many as 10% of the fuel pins in an assembly could exceed the average linear heat rate of 6.3 kw/ft, together with having a burnup of ≥54 GWD/MTU. For conservatism, it was assumed that 25% of the fuel pins in the damaged assembly exceed the RG 1.183 limits.

For the fuel pins that do not meet the operating limits identified in RG 1.183, the conservative gap fractions consistent with RG 1.25 (as modified by the direction of NUREG/CR-5009) will be used. These gap fractions are listed in Table 4.

Fission Product Form

In accordance with RG 1.183, the iodine species in the pool are 99.85% elemental and 0.15% organic. This is based on splitting the activity leaving the damaged fuel into 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodine. It assumed that all CsI is instantaneously dissociated in the water and re-evolves as elemental. Thus, 99.85% of the iodine released is elemental.

Pool Scrubbing Removal of Activity

For fuel pool water depths of 23 feet or greater above the fuel, RG 1.183 specifies an overall decontamination factor (DF) of 200. There is no retention of noble gases in the water.

The cesium and rubidium released from the damaged fuel rods are assumed to remain in a non-volatile form and would not be released from the pool.

Isolation and Filtration of Release Paths

Credit is not taken for removal of iodine by filters nor is credit taken for isolation of release paths.

The activity released from the damaged fuel assembly is assumed to be released to the outside environment at a uniform rate over a 2-hour period. Since no filters or containment isolation is modeled, this analysis supports refueling operation with the equipment hatch or personnel air lock remaining open.

The activities released from the water pool at 84 hours after shutdown are given in Table 6.

Control Room Isolation

The analysis assumes that the control room HVAC system is initially operating in normal mode. The activity level in the control room causes a high radiation signal within one minute. It is conservatively assumed that the radiation monitor that produces automatic isolation of the control room has failed. Subsequent manual isolation and switch to the control room HVAC emergency operation is assumed to take place 20 minutes after the high activity alarm.

Acceptance Criteria

The offsite dose limit at the EAB and LPZ for a fuel handling accident are 6.3 rem TEDE per RG 1.183. The limit for the control room dose is 5 rem TEDE from 10 CFR 50.67.

Results and Conclusions

The limiting fuel handling accident doses at 84 hours after shutdown with 1800 cfm unfiltered in-leakage modeled into the control room are:

Exclusion Area Boundary	4.0 rem TEDE
Low Population Zone	1.5 rem TEDE
Control Room	4.8 rem TEDE

All acceptance criteria are met.

The information presented in this report is applicable to Indian Point Unit 3 for Cycle 12. This information is applicable to future fuel cycles provided that the number of fuel pins exceeding the maximum linear heat generation rate of 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU does not exceed 25% of the total fuel pins per assembly.

The results of the analysis reported herein support fuel movement as early as 84 hours after shutdown without the use of containment purge isolation or filtration. The results also support removal of the technical specification 550-hour limit for fuel movement without containment purge isolation or aligning the system to discharge through the HEPA filters and charcoal absorbers, which when used, allow fuel movement at 145 hours (Reference 6).

References

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

2. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors", March 1972.

3. NUREG/CR-5009, Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors", February 1988.

4. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", EPA-520/1-88—0202, September 1988.

5. EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water and Soil", EPA 402-R-93-081, September 1993.

6. IP3 Technical Specifications
   Specifications Section 3.9 Refueling Operations, Subsection 3.9.3 Containment Penetrations
   Bases Section B.3.3 Instrumentation, Subsection B 3.3.6 Containment Purge System and Pressure Relief Line Isolation Instrumentation
   Bases Section B 3.9 Refueling Operations, Subsection B 3.9.3 Containment Penetrations, and Subsection B 3.9.6 Refueling Cavity Water Level

Table 1

Committed Effective Dose Equivalent (CEDE) Dose Conversion Factors
Effective Dose Equivalent (EDE) Dose Conversion Factors and
Nuclide Decay Constants

	CEDE	EDE	
Isotope	DCF (rem/curie)	DCF (rem·m^3/Ci·sec)	Decay Constant (hr^{-1})
I-131	3.29E4	6.734E-2	0.00359
I-132	3.81E2	0.4144	0.301
I-133	5.85E3	0.1088	0.0333
I-134	1.31E2	0.4810	0.791
I-135	1.23E3	0.2953	0.105
Kr-85m	N/A	2.768E-2	0.155
Kr-85	N/A	4.403E-4	7.38E-6
Kr-87	N/A	0.1524	0.545
Kr-88	N/A	0.3774	0.244
Xe-131m	N/A	1.439E-3	0.00243
Xe-133m	N/A	5.069E-3	0.0132
Xe-133	N/A	5.772E-3	0.00551
Xe-135m	N/A	7.548E-2	2.72
Xe-135	N/A	4.403E-2	0.0763
Xe-138	N/A	0.2135	2.93

Table 2

Offsite Breathing Rates and Atmospheric Dispersion Factors

	Offsite Breathing Rates (m^3/sec)
0 - 8 hours	3.5E-4

	Offsite Atmospheric Dispersion Factors (sec/m^3)
Exclusion Area Boundary 0 - 2 hours	1.03E-3
Low Population Zone 0 - 2 hours	3.8E-4

Table 3

Control Room Parameters

Volume (ft^3)	47,200
Normal Ventilation Flow Rates (cfm)	
Filtered Makeup Flow Rate	0.0
Filtered Recirculation Flow Rate	0.0
Unfiltered Makeup Flow Rate	1500
Unfiltered In-leakage Flow Rate	1800
Emergency Operation	
Filtered Makeup Air Flow Rate	400
Filtered Recirculation Flow Rate	1000
Unfiltered Makeup Air Flow Rate	0.0
Unfiltered In-leakage Flow Rate	1800
Filter Efficiencies (%)	
Elemental	90
Organic	90
Particulate	90
CR Radiation Monitor Sensitivity (μCi/cc)*	3.33E-4
Time to enter Emergency Operation HVAC mode after high radiation alarm.	20 minutes**
Breathing Rate - Duration of the Event (m^3/sec)	3.5E-4
Atmospheric Dispersion Factor (sec/m^3)	2.3E-3
Occupancy Factors	
0 - 24 hours	1.0

*Radiation monitor R33 checks an air sampling line drawing air from the control room bulk air. The monitor setpoint is based on a source with a 0.2 Mev / disintegration (similar to Xe-133).

** Operator action

Table 4

Assumptions Used for FHA Dose Analysis

Radial peaking factor	1.7
Fuel damaged (number of assemblies)	1
Time from shutdown before fuel movement (hr)	≥84
Activity in the damaged fuel assembly (Ci)	See Table 5
Minimum water depth	23 feet
Overall pool iodine scrubbing decontamination factor	200
Iodine chemical form in release to atmosphere (%) The split of the Iodine chemical form is based on an overall DF of 200	
Elemental	70
Organic	30
Particulate	0
Filter efficiency	No filtration assumed
Isolation of release	No isolation assumed
Time to release all activity (hours)	2
Gap Fractions	
<u>RG 1.183 Gap Fractions</u>	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
<u>RG 1.25 (as modified by the direction of NUREG/CR-5009)</u>	
I-131	0.12
Kr-85	0.30
Other iodines and noble gases	0.10

Table 5

Core Total Fission Product Activities
Based on 102% of 3025MWt
84 Hours After Shutdown

I-131	6.57E+07
I-132	6.03E+07
I-133	1.10E+07
I-134	1.09E-20
I-135	2.48E+04
Kr-85m	5.26E+01
Kr-85	1.04E+06
Kr-87	5.80E-13
Kr-88	7.68E-02
Xe-131m	9.33E+05
Xe-133m	2.70E+06
Xe-133	1.30E+08
Xe-135m	3.97E+03
Xe-135	7.46E+05
Xe-138	0.00E+00

Table 6

Fuel Handling Accident Activity Release
for an Accident Occurring 84 Hours Post Shutdown

I-131	2.60E+02
I-132	1.66E+02
I-133	3.03E+01
I-134	3.00E-26
I-135	6.83E-02
KR 85m	2.90E-02
Kr-85	1.37E+03
KR 87	3.19E-16
KR 88	4.23E-05
XE131m	5.14E+02
XE133m	1.49E+03
XE133	7.16E+04
XE135m	2.19E+00
XE135	4.11E+02
XE138	0.00E+00

ATTACHMENT IV TO IPN-02-044

Commitments for Implementation of Proposed Technical Specification Changes Regarding Use of Alternate Source Term for the Fuel Handling Accident

Commitment ID	Description
IPN-02-044-1	ENO will establish administrative controls to ensure prompt closure of containment openings in the event of a fuel handling accident in the containment building.
IPN-02-044-2	ENO will relocate the requirements of Technical Specifications 3.3.8 and 3.7.13 to the Technical Requirements Manual.