NUCLEAR ENERGY INSTITUTE

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July 27, 2006

Ms. Catherine Haney
Director, Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Code O-8 E1A
Washington, DC 20555-0001

**SUBJECT:** NEI 06-02, "License Amendment Request Guidelines," Final Draft for Comment, July 2006

**PROJECT NUMBER: 689**

Dear Ms. Haney:

This letter is addressed to you in your capacity as Chairman of the NRC Licensing Action Task Force (LATF).

In August 2001, NEI published a white paper entitled "Standard Format for Operating License Amendment Requests from Commercial Reactor Licensees." The objective was to encourage licensees to standardize the format and content of license amendment requests (LARs) as a means to improve the efficiency and effectiveness of the LAR process. Many licensees have incorporated the white paper into their administrative processes, and the white paper is referenced in NRR Office Instruction LIC-101, "License Amendment Review Procedures."

Based on industry experience with the white paper, NEI has upgraded it to a numbered guideline (NEI 06-02) and expanded its scope to include the use of precedent, the conduct of the "request for additional information" (RAI) process, and other administrative matters.

The guideline has received several rounds of industry review and is at the point where it would benefit from a final round of NRC and industry comments.

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Therefore, NEI requests that NRC review the final draft (enclosed) and provide comments to the undersigned by September 30, 2006. A similar request for final comments has been sent to NEI Administrative Points of Contact. When final comments have been resolved and incorporated, we plan to submit NEI 06-02 for NRC endorsement.

If you have questions or require additional information, please contact me at 202.739.8138 (jwr@nei.org) or Mike Schoppman at 202.739.8011 (mas@nei.org).

Sincerely,

Jack W Roe

Jack W. Roe

Enclosure

c: Mr. C. F. Holden, NRC
Ms. Michelle Honcharik, NRC
NEI Licensing Action Task Force Steering Group

# NEI 06-02

## LICENSE AMENDMENT REQUEST GUIDELINES

July 2006
[Draft for NRC Comment]

# Acknowledgements

NEI acknowledges the creative work of the NEI LAR Team and the many comments and insights provided by industry reviewers and the Licensing Action Task Force (LATF) during preparation of this Guideline.

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# ABSTRACT

Several administrative steps are associated with the licensee submittals and NRC staff reviews required to amend an operating license for a commercial nuclear power plant. The steps are:

1. Initiating the License Amendment Process – a licensee initiates the license amendment process (pursuant to 10 CFR 50.90) whenever it determines that a proposed activity (e.g., plant modification, procedure change) requires modification of the plant Operating License or Technical Specifications. The License Amendment Request (LAR) process can be initiated by the licensee pursuant to the regulations (e.g., 10 CFR 50.59) or by NRC direction (e.g., plant-specific implementation of a generic requirement).

2. Use of Precedent – a licensee seeking regulatory approval to conduct a proposed activity should identify relevant precedent-setting license amendments on its own docket or on other dockets to support the acceptability of the proposed activity.

3. Standard Format for LARs – a licensee has the option to prepare a license amendment request using the format and content guidance contained in NEI 06-02.

4. Licensee Interface with NRC – a licensee communicates with the NRC staff as necessary to facilitate:
    - Pre-submittal communications and meetings
    - NRC work planning
    - Public notification (Federal Register) of proposed licensing action
    - NRC review (acceptance review, RAIs, meetings)
    - Supplements to the initial LAR
    - NRC issuance (or rejection, or request for withdrawal)

5. Documentation – A license amendment request may be approved or rejected by NRC, or withdrawn by the licensee. Approved amendments are followed by licensee implementation. Rejected amendments may be appealed or resubmitted in revised form. Withdrawn amendments may be tabled or resubmitted at some future time. In any case, the outcome should be documented for future reference.

6. Resolution of Disagreements – a licensee has recourse to administrative processes to request formal resolution of disagreements with the NRC.

NEI 06-02 describes a standardized process that licensees and the NRC staff may use on a voluntary basis to guide the administrative interface during the LAR process. The objective is to improve the efficiency and effectiveness of the licensee's preparation and the NRC staff's review by describing key administrative steps associated with the preparation and review of LARs, such as the use of precedent, the use of standard format and content guidance for both industry LARs and NRC SEs, and the resolution of disagreements that may arise during the process. It is not intended to be a guideline that licensees can use for the technical development, validation, review and approval of LARs.

NEI 06-02 has been endorsed by NRC in *[placeholder for citation of NRC reference document].*

**Disclaimer – Discussions of NRC activities in NEI 06-02 are illustrative and are not binding on the NRC staff. In all cases, NRC activities are controlled by NRC internal guidance**

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# 1.0 INITIATING THE LICENSE AMENDMENT PROCESS

The holder of an Operating License (OL) for a commercial nuclear power plant (the "licensee") uses administrative means to manage change (e.g., plant modifications, procedure changes, program changes, etc.). Most changes are controlled by the licensee, but NRC review and approval is required in some cases.

The licensee evaluates planned changes to determine whether prior NRC review is necessary. In this context, "change" is defined by 10 CFR 50.59 *[Reference 1]* as "a modification or addition to, or removal from, the facility or procedures that affects a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished." Detailed guidance on the implementation of 10 CFR 50.59 is contained in NEI 96-07 *[Reference 2]*, which has been endorsed by NRC in Regulatory Guide 1.187 *[Reference 3]*.

In some cases, regulations other than 10 CFR 50.59 establish the criteria for determining whether NRC review is necessary, for example:

1. 10 CFR 50.12 *[Reference 4]* specifies a process that may be used as an alternative to the license amendment process to apply for an exemption from the requirements of a specific regulation.

2. 10 CFR 50.46(a)(3) *[Reference 5]* specifies an alternative change process to 10 CFR 50.59(c)(2)(viii) for LOCA evaluation models.

3. 10 CFR 50.54(a) *[Reference 6]* specifies the change process for the quality assurance program.

4. 10 CFR 50.54(p) *[Reference 7]* specifies the change process for the security plan and the guard training and qualification plan.

5. 10 CFR 50.54(q) *[Reference 8]* specifies the change process for emergency plans.

6. 10 CFR 50.55a *[Reference 9]* specifies the processes for requesting alternatives to, or relief from, the inservice inspection and testing requirements of the ASME Code.

7. 10 CFR 50.65 *[Reference 10]* specifies the maintenance program requirements for monitoring the performance of SSCs compared to licensee-established goals.

Change-control processes are discussed in a number of licensee programs, industry guidelines, and NRR Office Instructions, for example :

1. Technical requirements manual (repository for information outside the UFSAR).

2. Technical Specification Bases Control Program.

3. NEI 97-04 *[Reference 11]*, endorsed by NRC in Regulatory Guide 1.186 *[Reference 12]* (guidance for evaluating and dispositioning design discrepancies).

4. NEI 98-03 *[Reference 13]*, endorsed by NRC in Regulatory Guide 1.181 *[Reference 14]* (guidance for periodically updating the UFSAR).

5. NEI 99-04 *[Reference 15]*, endorsed by NRC in SECY-00-0045 *[Reference 16]*, contains guidance for changing licensee commitments made to NRC in docketed correspondence.

6. NEI 01-01, revision 1 *[Reference 17]*, endorsed by NRC in RIS 2002-22 *[Reference 18]* (guidance for performing 10 CFR 50.59 evaluations associated with the introduction of digital technology).

7. NRR Office Instruction LIC-100 *[Reference 19]* (NRC guidance on the terminology and documents associated with the licensing bases for an operating nuclear power plant).

8. NRC Office Instruction LIC-101 *[Reference 20]* (NRC guidance on the license amendment review process).

If the licensee's evaluation concludes that NRC approval must be obtained in the form of an amendment to the OL, the licensee must submit an LAR to the NRC in accordance with 10 CFR 50.90 *[Reference 21]*.

## 2.0 USE OF PRECEDENT

Precedent-setting licensing actions can help reduce the licensee's developmental effort, reduce the need for NRC RAIs, and lead to a more predictable and abbreviated regulatory review schedule. However, limitations accompany the use of precedent. The licensee has the primary burden to identify relevant LARs on other dockets, Topical Reports, NRC Safety Evaluations (SEs), and other precedent-setting correspondence and regulatory action. Differences between the licensee's LAR and the referenced precedent(s) must be identified and dispositioned as acceptable or not applicable.

The effective use of precedent has three main components:

1. Access to NRC-approved, precedent-setting documents.

2. Licensee submittal of an LAR that provides appropriate justification based on the precedent SE and supporting documents (i.e., applicability, differences, acceptability of differences, etc.).

3. NRC acceptance that the proposed precedent would improve the efficiency of the regulatory review and help establish a more predictable review schedule.

## 2.1 Sources of Precedent

The following are possible sources for identifying NRC-approved, precedent-setting amendments, Topical Reports, and NRC SEs:

1. Plant-specific experience with similar amendments

2. Information search services

3. Nuclear industry groups:
   a. NEI, INPO, and EPRI
   b. NSSS Vendors and Owners Groups
   c. Regional utility groups
   d. STARS (Strategic Teaming and Resource Sharing) group
   e. USA (Utilities Service Alliance) group
   f. Standards organizations
   g. Groups formed in response to a particular technical issue
   h. Ad hoc communication among licensees

4. Government sources:
   a. NRC Agency Documents Access and Management Systems (ADAMS)
   b. NRC Public Document Room
   c. Federal Register

5. Topical Reports for which NRC has issued generic safety evaluations

6. Group/Fleet submittals approved by NRC as a common basis for a group or fleet of plants

7. TSTF Travelers approved by NRC (as discussed in Appendix C)

8. Risk-Informed licensing actions as discussed in LIC-101

9. CLIIP notices as discussed in LIC-101

The most effective precedent is a generic "model safety evaluation" developed pursuant to the Consolidated Line Item Improvement Process (CLIIP) [Reference 22]. NRC publishes proposed CLIIPs in the Federal Register for public comment. If public comments are satisfactorily resolved, NRC publishes a final model SE in the Federal Register for licensees to reference as the basis for plant-specific LARs.

Technical Specification Task Force (TSTF) Travelers are also effective precedents. The TSTF is an Owners Group task force that develops proposed changes to the improved Standard Specifications (ISTS) [Reference 23].

In addition, NEI maintains a key-worded spreadsheet that lists all completed 10 CFR 50 licensing actions beginning in calendar year 2000, including references to the corresponding LARs and supplemental letters. The spreadsheet is accessible to member companies on the password-protected NEI LATF website (http://www.latf.net/50-090/RAI/Spreadsheet-r5.xls).

## 2.2 Applicability of Precedent

NRC internal guidance on the use of precedent is contained in NRR Office Instruction LIC-101. It defines precedent licensing actions as those with a similar proposed change and regulatory basis. The use of precedent in the regulatory review process increases the efficiency of the review, minimizes the need for requests for additional information (RAIs), and improves the consistency among similar licensing actions.

An LAR that relies on precedent outside the CLIIP should reference the affected power plant(s) and amendment number(s), and discuss how the precedent applies to the specific circumstances of the proposed amendment. Precedent, by itself, does not demonstrate the acceptability of a proposed amendment, but it does give the NRC information about how the agency has treated similar changes in the past.

## 2.3 Licensee Treatment of Precedent

The following considerations relate to the identification and use of precedent:

1. The use of precedent is voluntary. However, the NRC recognizes that there are significant efficiencies to be gained by using applicable precedent, especially for LARs that are first-of-a-kind (FOAK), technically complex, or based on a generic topical report. Therefore, a useful early step in preparing a LAR is to identify, assess, and review potential precedents.

2. Typically, a single precedent is sufficient, although licensees may cite multiple precedents.

3. The licensee has the primary responsibility to identify precedent.

4. The NRC may, but is not obligated to, identify potential precedents. If so, it is the licensee's obligation to ensure that the proposed precedent is appropriate.

5. Contact the precedent-setting licensee to verify that the relevant SSCs in the precedent plant are sufficiently similar to those addressed in the LAR.

6. Regardless of the precedent source, discuss how the precedent applies to the LAR. Look for consistency with respect to:
   a. physical characteristics
   b. design basis
   c. risk-significance
   d. scope and depth of technical justification

7. The precedent-seeking licensee has the obligation to perform a thorough design/licensing basis comparison to verify that the proposed precedent is appropriate for use in the proposed amendment. Identify and justify all differences between the precedent and the LAR that are relevant to the issue being addressed by the proposed amendment. Even if the LAR closely follows the precedent, associated SSCs may be sufficiently different that the proposed precedent might not apply either in whole or in part. For example, the plant-specific TS could be the same (or similar) as the precedent TS, but the design bases, licensing bases, and UFSAR documentation could be different.

8. Include a summary of the precedent(s) in the submittal. The summary should include:
   a. a discussion of how the precedent(s) applies to the LAR
   b. a discussion of the differences between the licensee's plant and the precedent plant(s) that are relevant to the scope of the LAR
   c. references to all precedent-related documents, e.g., LARs, LAR supplements, RAIs, NRC SEs, etc.

9. Communicate the proposed use of precedent to the NRC Project Manager (PM) early in the development of the LAR. NRC PMs treat informal communications as a routine job function, and can facilitate and expedite the exchange of information with technical reviewers.

10. Request pre-submittal discussions with NRC if that would be useful in determining Staff expectations with respect to scope, format and technical content. This step is advisable if the submittal is technically complex or FOAK. The pre-submittal interface could facilitate followup action, e.g., expand the precedent search, withdraw the amendment request, or resolve NRC staff concerns. NRC would need to establish administrative control to collect labor hours for pre-submittal interactions.

11. Take advantage of electronic bulletin boards and internet websites to expedite the exchange of information with NRC about plant-specific LAR reviews. For example, NRC public websites have been used to facilitate license renewal and TS conversion projects.

12. Provide feedback to NEI regarding precedent experience so NEI can update the license amendment spreadsheet.

### 2.4 NRC Treatment of Precedent

Guidelines for NRC staff review of license amendment requests are contained in NRR Office Instruction LIC-101. The primary objectives of LIC-101 are (1) consistent processing of license amendments, and (2) technical consistency between similar amendments. An important step in meeting these objectives is the appropriate use of precedent set by prior, similar licensing actions.

Precedent documents can be a valuable input to the NRC work plan and the SE. They can help the PM and technical branches develop a review plan, identify a lead reviewer, avoid duplication of past RAIs, and reduce the overall resources necessary to complete the review.

LIC-101 provides that proposed precedent be reviewed for applicability, accuracy, and completeness when compared with the incoming LAR and its associated plant-specific design details. The staff verifies that the precedent is appropriate for use with LAR and meets current NRC expectations with respect to format, content, guidance, and findings.

## 3.0 STANDARD FORMAT FOR LICENSE AMENDMENT REQUESTS

Appendix A to this guideline supersedes the guidance published by NEI in October 2002 [Reference 24].

Appendix A outlines a standardized format that licensees may use on a voluntary basis to prepare a proposed plant-specific request for an amendment to the Operating License for a commercial reactor. The standard LAR package includes a cover letter with the following Enclosures and Attachments:

1. Enclosure 1 to the Cover Letter – Licensee Evaluation (technical and regulatory evaluation of the proposed amendment)

2. Required Attachment 1 to the licensee evaluation in Enclosure 1 – TS Page Markups

3. Optional Attachment 2 to the licensee evaluation in Enclosure 1 – Changes to TS Bases (identify the changes needed to bring the Bases into conformance with the proposed TS changes)

4. Optional Attachment 3 to the licensee evaluation in Enclosure 1 – Retyped TS Pages

5. Enclosure 2 to the Cover Letter – List of Regulatory Commitments (formal commitments associated with the proposed amendment)

Italicized information in brackets represents amendment-specific information to be inserted by the licensee.

Footnotes are used to explain certain concepts. Thus, they are part of NEI 06-02, not part of the LAR format.

# 4.0 LICENSEE INTERFACE WITH NRC

NRR Office Instruction LIC-101 describes the overall process used by the NRC staff to conduct LAR reviews. The basic steps are work planning, public notice and comment, safety evaluation, regulatory evaluation, and documenting results.

With respect to work planning, the LAR acceptance review and the NRC search for precedent are the steps of interest to licensees. The NRC project manager, with technical branch assistance if necessary, reviews the LAR for completeness. The requirements and key elements of the acceptance review are described in LIC-101.

With respect to the public notice and comment process, the Federal Register notice and the resolution of public comments are the steps of interest to licensees. The process is described in LIC-101.

With respect to the conduct and documentation of the NRC safety and regulatory evaluations, the steps of interest to licensees are the treatment of precedent, the review of licensee commitments, and the RAI process. The use of precedent is discussed above in Section 2 and in LIC-101. The commitment process is discussed in NEI 99-04 and in LIC-101. The remainder of this section is supplemental guidance with respect to the RAI process.

## 4.1 Overview of the RAI Process

As a general rule, a quality LAR will contain sufficient information for the NRC to complete its review without requesting additional information. However, if the NRC determines that additional information is needed to support the regulatory review of a licensee's plant-specific LAR, it prepares a "request-for-additional-information" (RAI) for transmittal to the licensee.

The NRC uses the RAI process when information it believes is necessary is not included in the initial LAR, is not contained in any other docketed correspondence, or cannot reasonably be inferred from other sources of information readily accessible by the NRC staff. Frequent and early communication between the NRC Project Manager, the NRC technical staff, and the licensee can minimize the need for RAIs.

Informal communication (e.g., telecon or e-mail) is an expeditious means of requesting and providing explanatory information to expedite the NRC review. However, an informal RAI should be limited to the specifics of the LAR, and the response should not involve significant effort on the part of the licensee. The licensee has the option to ask NRC to convert an informal RAI into a formal RAI. Similarly, the licensee has the option to provide a formal response to an informal RAI. The objective is to expedite the regulatory review of the LAR.

Information that will form part of the basis for the reviewer's conclusion is submitted formally under oath and affirmation so NRC can incorporate it into the licensee's docket file in the NRC Public Document Room and in ADAMS.

Some of the factors that affect the RAI process are:

- submittal quality
- submittal complexity
- access to background information
- mutual understanding of objectives and expectations
- mutual agreement on submittal scope and level of detail
- treatment of the current licensing basis (CLB)
- depth of acceptance review
- personnel changes
- use of standardization
- management oversight

## 4.2 Steps in the RAI Process

RAIs may be informal (i.e., undocketed verbal or e-mail exchanges) or formal (i.e., docketed letters or e-mail correspondence). Typically, the steps in the RAI process are as follows:

1. NRC technical reviewers determine a need for additional information and draft RAIs for NRC management review. Licensees and NRC reviewers are encouraged to categorize individual questions in accordance with Appendix C. The informal use of a standard set of categories can help clarify the regulatory basis of each question and aid the licensee in preparing a concise but thorough answer.

2. The cognizant NRC technical branch chiefs or section chiefs review the draft RAIs for concurrence.

3. The cognizant NRC PM and branch chief in the Division of Operating Reactor Licensing review the draft RAIs for consistency with NRR Office Instruction LIC-101.

4. The NRC PM has the option to forward the draft RAIs to the licensee by e-mail.

5. The NRC PM and the licensee schedule a telecon to discuss the draft RAIs.

6. The licensee reviews the draft RAIs in preparation for the telecon with NRC. Licensees are encouraged to categorize individual questions in accordance with Appendix C.

7. An NRC/licensee telecon is conducted to:
   a. compare the NRC categorization of the questions with the licensee categorization (optional),
   b. ensure mutual understanding of what is being requested, or clarify questions if necessary,
   c. eliminate RAI questions for which the licensee can provide the requested information during the telecon such that formal correspondence to docket the information is not necessary,
   d. eliminate RAI questions for which NRC agrees the information is not needed,
   e. agree on a date by when the licensee can provide supplemental information to RAI questions that were not resolved during the telecon, and
   f. identify disputed questions for followup action.

8. NRC may request a supplemental letter from the licensee to document certain information provided during the telecon. The NRC PM and the licensee should agree on a target response date.

9. NRC may supplement its informal RAI with a formal RAI letter. The licensee also may request a formal RAI letter.

10. If an RAI question results in a change to the LAR, the licensee should provide sufficient supplemental information to fully explain the nature, context, and basis for the change.

## 4.3 <u>Licensee Checklist for Minimizing RAIs</u>

1. Consider the point of view of the NRC reviewer when drafting an LAR.

2. Use language that NRC can use in the SE.

3. Use the NEI LAR spreadsheet and other information resources to look for similar LARs on other dockets.

4. Review precedent LARs and associated references, and include relevant information in the new LAR.

5. Use the RAI categorization process described in Section 4.3 to identify deficiencies in LAR preparation (licensee deficiencies) and deficiencies in RAI preparation (NRC deficiencies).

6. For first-of-a-kind and complex LARs, consider scheduling a pre-submittal meeting with the NRC staff. Pre-submittal meetings can clarify the licensee's objectives in submitting the LAR and enhance the licensee's understanding of the NRC's point of view. The objective of a pre-submittal meeting is a mutually acceptable regulatory review schedule.

7. Optimize the use of telecons to discuss draft RAIs with the NRR PM and applicable technical reviewers. Make a record of interpretations and agreements reached during telecons. Share notes with NRC after the call to ensure common understanding going forward. Maximize the use of informal means to disposition RAIs.

8. Use a clear format to respond to questions. A recommended format is to repeat the question in its entirety and then to provide the licensee response.

9. After completing a licensing action, conduct a lessons-learned debriefing to identify ways to improve the overall LAR process.

### 4.4 NRC Treatment of Generic Issues

In the context of NEI 06-02, a "generic RAI" is a question posed during the NRC review of a plant-specific LAR that refers to an agency position on a generic issue that, in the reviewer's opinion, must be incorporated into the review of the LAR. If a licensee receives what it believes is an inappropriate generic RAI, it should forward the question to the NEI Licensing Action Task Force (LATF) and the NRC LATF for dispositioning.

The LATF will discuss the relevance of the question to the plant's CLB, as defined in 10 CFR 54.3(a) [Reference 25] and in NRC guidance on the resolution of degraded and nonconforming conditions [Reference

26]. NRC should not impose generic staff positions during review of plant-specific LARs, unless such imposition is necessary in response to an immediate plant-specific safety or compliance concern. Otherwise, conformance with the CLB is a sufficient (and necessary) basis for NRC approval of the LAR. The generic communication process should be used to implement NRC staff recommendations derived from the resolution of generic issues.

# 5.0 DOCUMENTATION OF RESULTS

## 5.1 NRC Approval and Licensee Implementation

Upon receipt of an NRC-approved license amendment, the licensee should review the amendment and attached Safety Evaluation (SE). If incorrect or incomplete information is identified in the SE, the licensee should document the concerns and promptly inform the NRR PM. The licensee should communicate with the PM (e.g., conference calls and email) to confirm the bases for the amendment and to document the confirmation in plant records. For errors that conflict with the bases, the licensee should request a revised SE from the NRC.

The licensee should confirm the acceptability of the implementation date of the amendment. Plans should be established, if not already in place, for implementation consistent with the requirements of the approved amendment. Revised implementation dates should be communicated to the NRC PM to determine if a supplement to the license amendment is necessary.

## 5.2 Licensee Checklist for Resolution of Disagreements with NRC

In some cases, NRC may disagree with a licensee's determination that a proposed action does not require a license amendment, and may request that a LAR be submitted and approved before the licensee implements the proposed action. In other cases, the NRC may respond to a LAR by rejecting it during the acceptance review, by requesting additional information during the review, or by denying it upon completion of the review. If a licensee disagrees with the NRC response to a proposed action, it has recourse to the following steps to determine whether the NRC position is consistent with 10 CFR 50.109 [Reference 27] (i.e., is not a new or different staff position) or is otherwise justified:

1. Refer to NRR Office Instruction LIC-101 for a description of the NRC process for reviewing LARs.

2. Query other licensees to determine if they have experienced a similar disagreement.

3. Determine if an NRC Regional Office is involved. If so, the NRC position may derive from a Task Interface Agreement (TIA) between the Region and NRC Headquarters. Refer to NRR Office Instruction COM-106 [Reference 28] for internal NRC guidance on the use of TIAs to gather information about plant-specific licensing bases, regulatory requirements, technical positions, plant configurations, or operating practices in support of NRC review of an issue, event, or inspection finding.

4. Initiate informal discussions with NRC (telecon, e-mail).

5. If warranted, escalate the formality of the process. Engage industry and NRC management in the regulatory dialogue. Consider the following options:

   a. Request a plant-specific meeting with NRC to discuss the disagreement. Document the expectations, interpretations, and factual information discussed during the meeting. Prepare a joint resolution plan and schedule.

   b. Request generic resolution through the NRC/NEI Licensing Action Task Force (LATF) interface. The periodic public meetings between the NRC LATF and the NEI LATF are a forum for raising industry concerns with the regulatory use of preliminary generic information. Absent an immediate plant-specific safety concern or non-compliance, a generic resolution process (rather than the plant-specific LAR process) is the preferred pathway to resolve issues that apply to all PWRs and/or BWRs, or to a significant subset of PWRs/BWRs. A front-loaded technical review leading to a generic safety evaluation (SE) leads to a more efficient and effective resolution than a series of separate and evolving plant-specific SEs.

6. Request interpretation by the NRC Office of the General Counsel (OGC). Official NRC interpretations are limited to those contained in documents reviewed by, or statements made by, OGC.

7. Request an NRC regulatory analysis pursuant to NUREG/BR-0058 [Reference 29].

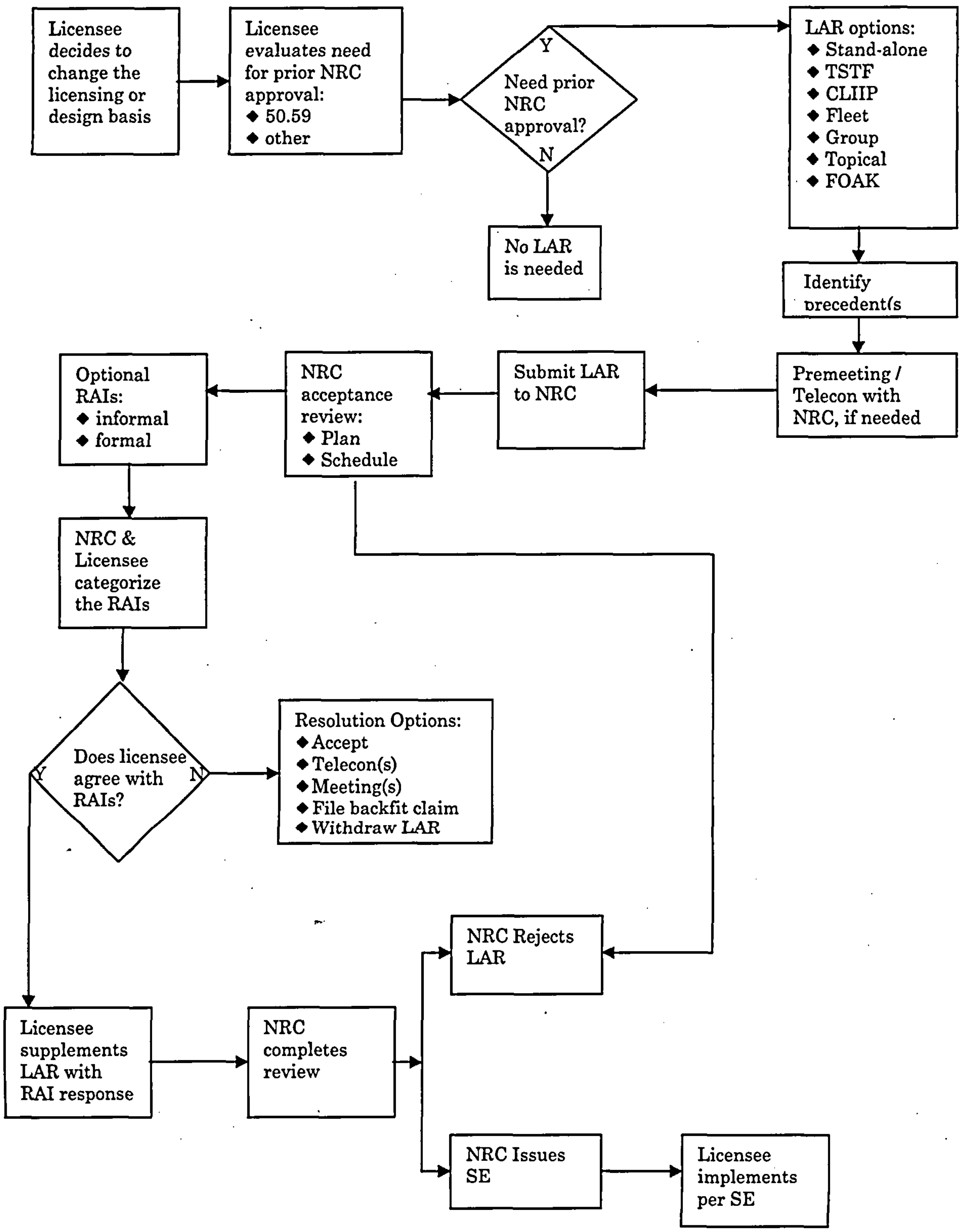
8. File a backfit claim or appeal pursuant to 10 CFR 50.109. Refer to NRR Operating Instructions LIC-202 [Reference 30] and LIC-400 [Reference 31] for additional regulatory guidance on plant-specific backfits and generic requirements, respectively.

9. File a petition for rulemaking, if applicable.

10. Request a hearing pursuant to 10 CFR 2 [Reference 32].

11. Seek judicial remedy through the courts.

## FIGURE 4-1 – LAR Flow Chart



## Standard Format for License Amendment Requests from Operating Reactor Licensees

Appendix A outlines a standardized format that licensees may use on a voluntary basis to prepare a proposed plant-specific request for an amendment to the Operating License for a commercial reactor. The standard package described in this Appendix includes the following:

1. Licensee's cover letter briefly describing the objectives and bases of the proposed amendment.

2. Licensee's evaluation (technical and regulatory) of the proposed amendment.

3. Technical Specification page markups showing the proposed changes.

4. Optional Technical Specification Bases page markups showing the changes that would be needed to bring the Bases into conformance with the proposed TS changes.

5. Optional Retyped Technical Specification pages.

6. A list of Regulatory Commitments associated with the proposed amendment.

Italicized information in brackets represents amendment-specific information to be inserted by the licensee.

Footnotes are used to explain certain concepts. Thus, they are part of this guideline, not part of the LAR format.

*[Licensee's letterhead]*

*[Date]*

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: *[Plant / Unit Name(s)]*
*Docket No(s) [50-___, 50-___]*
*[Title]*[1]

Pursuant to 10 CFR 50.90, *[license holder]* hereby requests the following amendment: *[Include a brief summary of the proposed amendment and the results of the corresponding "significant hazards determination." If the proposed amendment is consistent with a Technical Specification Task Force (TSTF) change to the Standard Technical Specifications (STS), include a statement to that effect, and provide a reference to the applicable TSTF Traveler number (TSTF-xxx) and title.]*. *[License holder]* requests approval of the proposed amendment by *[date + justification]*.[2] Once approved, the amendment shall be implemented within *[ ]* days.[3]

*[Include or attach a listing of formal licensee commitments that would derive from NRC's approval of the proposed amendment.]*

[1] The title used by many licensees is "License Amendment Request (LAR)." Other licensees use "Proposed License Amendment (PLA)." These and other equivalent terms are acceptable titles.

[2] Provide justification in the cover letter for the "need date." For example, if approval by that date is necessary to prepare for startup after a refueling outage.

[3] A 60-120 day implementation period is typical. If additional implementation time is needed, provide justification in the cover letter, e.g., if significant procedure changes are necessary to support implementation, or if significant plant modifications require a refueling outage for installation.

*[In accordance with 10 CFR 50.30(b), a license amendment request must be executed in a signed original under oath or affirmation. This can be accomplished by attaching a notarized affidavit confirming the signature authority of the signatory, or by including the following statement in the cover letter: "I declare under penalty of perjury that the foregoing is true and correct. Executed on (date)." The alternative statement is pursuant to 28 USC 1746. It does not require notarization.]*

If you have any questions or require additional information, please contact *[Mr./Mrs./Ms., licensee's point of contact for the NRC Office of Nuclear Reactor Regulation]* at *[telephone number]*.

Sincerely,

*[Signature]*

*[Name]*
*[Title]*

Enclosures:
1. Licensee Evaluation
   - Attachment 1 – TS Page Markups
   - Attachment 2 – Changes to TS Bases (optional)
   - Attachment 3 – Retyped TS Pages[4] (optional)
2. List of Commitments

cc: *[Region_]*
*[NRR Project Manager]*
*[Resident Inspector(s)]*
*[State contact]*

[4] Retyped pages may be submitted with the license amendment request, or they may be deferred until the end of the process to accommodate revisions derived from responses to NRC Requests for Additional Information or other sources.

# LICENSEE EVALUATION

Subject: *[Brief title. Identify which Technical Specification section(s) will be changed.]*

1.0 SUMMARY DESCRIPTION

2.0 DETAILED DESCRIPTION

3.0 TECHNICAL EVALUATION

4.0 REGULATORY EVALUATION

4.1 Significant Hazards Consideration

4.2 Applicable Regulatory Requirements/Criteria

4.3 Precedent

4.4 Conclusions

5.0 ENVIRONMENTAL CONSIDERATION

6.0 REFERENCES

## 1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License(s) *[license number(s)]* for *[plant/unit name(s)]*.

The proposed change(s) would revise the Operating License(s) to *[describe the proposed amendment, the reason for the amendment, and any timing constraints. Reserve details for Section 2.0.*[5] *If the proposed change is based on a TSTF Traveler and there are differences between the proposed change and the Traveler, identify and justify the differences.]*

## 2.0 DETAILED DESCRIPTION

*[Include:*

- *System description(s).*
- *Applicable references to UFSAR text and figures.*
- *Discussion of conditions that the proposed amendment is intended to resolve.*
- *An explanation of the circumstances that establish a need for the proposed amendments(s), for example, historical information, prior communication, or correspondence with NRC staff, relevant reference documents, etc.*

## 3.0 TECHNICAL EVALUATION

*[Include:*

- *A detailed explanation of the proposed amendment.*
- *A detailed description of analytical methods, applicable standards, data, and results.*
- *Technical details in support of safety arguments.*
- *The impact on UFSAR accident analyses.*
- *A discussion of relevant precedents.*
- *Briefly summarize the preceding arguments at the end of this section.]*

*[If the proposed amendment is risk-informed, include information in accordance with the Regulatory Guide series 1.174- 1.178 on "risk-informed decision-making." These five Regulatory Guides address plant-specific changes to the licensing basis, inservice testing, graded Quality Assurance, Technical Specifications, and inservice inspection, respectively).]*

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[5] In some cases, the amendment will affect only the Operating License. In most cases, the amendment also will affect one or more Technical Specifications.

*[The Technical Evaluation section should be written such that it may be used with minimal modification in the NRC staff's Safety Evaluation (SE).]*

## 4.0 REGULATORY EVALUATION

*[Provide a paragraph containing a few descriptive sentences suitable for use by NRC in the Federal Register notice that will be published to seek public comment on the proposed amendment. Avoid slang words or undefined abbreviations or acronyms. This summary may duplicate wording in the licensee's cover letter and should bound the detailed changes being proposed.]*

### 4.1 Significant Hazards Consideration[6]

*[Licensee name]* has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   *[For guidance on preparing a basis for this response, see the First Standard from RIS 2001-22: "Consider the effect of the change on structures, systems, and components (SSCs) of the plant to determine how the proposed change affects plant operations, any design function or an analysis that verifies the capability of an SSC to perform a design function. Determine if the proposed amendment would change any of the previously evaluated accidents in the UFSAR. The word "accidents" refers to anticipated (or abnormal) operational transients and postulated design basis accidents, including the events with which the plant must be able to cope (e.g., earthquake, flooding, turbine missiles, and fire) as described in the UFSAR. Determine if SSCs, operating procedures, and administrative controls that are affected have the function of preventing or mitigating any of these accidents. If the proposed change increases the likelihood of the malfunction of an SSC,*

[6] General guidance is contained in NRC Regulatory Issue Summary 2001-22, "Attributes of a Proposed No Significant Hazards Consideration."

*the potential impact on analyzed accidents should be considered (e.g., an increased likelihood of an SSC malfunction may increase the probability or consequences of an accident). If there is no impact on previously evaluated accidents, explain why."*

*"Discuss the differences in the probability and consequences of these accidents (or the bounding scenario) before and after the change and whether the differences are significant. If the change is not considered significant, explain why. Whether an increase is significant should be assessed case-by-case. A qualitative judgment may need to be made. Values of probability or consequence that continue to meet the licensing basis or applicable guidelines in the Standard Review Plan are generally not considered significant changes. If the probability of occurrence remains within the ranges already presented in the UFSAR for initiating events, then the increase is not considered significant. An increase beyond any of these values that is not deemed significant should be justified. The significance determination should include a comparison of the value before the change to that after the change. A large increase might not be considered significant in one situation, but a relatively small increase might be significant in another situation. The licensee should adequately justify the proposed determination."]*

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

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

   Response: No.

   *[For guidance on preparing a basis for this response, see the Second Standard from RIS 2001-22: "Determine whether the proposed amendment will change the design function or operation of the SSCs involved, or whether interim processes (e.g., process of installing a new system component or construction of a new facility, performance of testing or maintenance) will affect the SSCs' operation or its ability to perform its design function. Then determine whether the proposed change will create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases. This new accident would have been considered a design basis accident in the UFSAR had it been previously identified. A new initiator of the same accident is not a different type of accident. Finally, the accident must*

*be credible within the range of assumptions previously applied (e.g., random single failure, loss of off-site power, no reliance on non-safety-grade equipment)."]*

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 amendment involve a significant reduction in a margin of safety?

Response: No.

*[For guidance on preparing a basis for this response, see the Third Standard from RIS 2001-22: "Safety margins are applied at many levels to the design and licensing basis functions and to the controlling values of parameters to account for various uncertainties and to avoid exceeding regulatory or licensing limits. The specific values that define margin are established in each plant's licensing basis. Licensees should identify the safety margins that may be affected by the proposed change and review the conservatism in the evaluation and analysis methods that are used to demonstrate compliance with regulatory and licensing requirements"*

*"The safety margin before the change should be compared to the margin after the proposed change to determine if the amendment will reduce the margin, and if the change is significant. If a change does not exceed or alter a design basis or safety limit (i.e., the controlling numerical value for a parameter established in the UFSAR or the license) it does not significantly reduce the margin of safety. In other cases, the assessment of significance for this standard should be made on the same basis as discussed in the guidance for the first standard. Uncertainties and errors need to be considered in calculating the margin."]*

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

Based on the above, *[licensee name]* concludes that the proposed amendment(s) does (do) not involve a 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.

4.2 Applicable Regulatory Requirements/Criteria

*[This section describes in detail how the licensee's technical analysis, which may or may not include risk information, satisfies all applicable regulatory requirements and criteria. Any formal commitments to administrative controls needed to ensure compliance should be included in this section. The Regulatory Analysis provides a basis that the NRC staff may use to find the proposed amendment acceptable. It should be written such that it may be used with minimal modification in the NRC staff's Safety Evaluation (SE).] [To assist the NRC staff, the licensee may choose to include an optional table of applicable regulatory requirements/criteria.]*

### 4.3 Precedent

*[If precedent can be identified, the licensee should reference the affected power plant(s) and amendment number(s), and briefly discuss how the precedent applies to the specific circumstances of the proposed amendment. If there are any differences between identified precedent and the proposed amendment, the licensee should explain the differences and describe their impact on the acceptability of the proposed amendment. Precedent, by itself, does not demonstrate the acceptability of a proposed amendment, but it does give the NRC staff information about how they have treated similar changes in the past. This may simplify the NRC staff's review.]*

### 4.4 Conclusions

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

## 5.0 ENVIRONMENTAL CONSIDERATION

*[The identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review is the subject of 10 CFR 51.22. The categories of actions deemed "categorical exclusions" are specified by 10 CFR 51.22(c). The licensee's consideration of environmental factors should include sufficient detail to support a finding of categorical exclusion. For the majority of changes, it is clear that the environment will not be affected (e.g., extending a surveillance interval). Therefore, a simple statement (see below) is sufficient. If appropriate, the licensee can provide more detailed information to strengthen the justification of categorical exclusion.]*

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, 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 effluents 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.

[and/or]

The proposed amendment is confined to (i) changes to surety, insurance, and/or indemnity requirements, or (ii) changes to recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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 REFERENCES

*[Identify and number all references used to prepare the proposed amendment. Each reference should be cited at least once in this Enclosure (Licensee's Evaluation). If a reference is needed to understand, review, or approve the proposed amendment, it should be considered for inclusion as an attachment and identified with a suitable attachment number or letter.]*

# TS PAGE MARKUPS

*[Mark up affected Technical Specification pages by either of the following methods:*

1. *Word-processor mark-ups using the program's "redline/strikeout" feature*
2. *Hand-written mark-ups of copies of the affected pages]*

## CHANGES TO TS BASES

*[Mark up affected Technical Specification Bases pages by either of the following methods:*

1. *Word-processor mark-ups using the program's "redline/strikeout" feature*
2. *Hand-written mark-ups of copies of the affected pages]*

# RETYPED TS PAGES

*[Re-type the affected Technical Specification pages to incorporate the proposed changes]*

## LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by *[Licensee]* in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to *[name of licensee contact].*

| REGULATORY COMMITMENTS |
|---|
| *[1. Duplicate the commitment wording from the body of the LAR cover letter. If committing to complete an action by a specific date, include the date in the cover letter and in this table. Guidance on controlling regulatory commitments is contained in NEI 99-04 and NRR Office Instruction LIC-105.]* |
| *[2.]* |
| *[3.]* |
| *[4.]* |

## SAMPLE LICENSE AMENDMENT REQUEST

This Appendix provides a sample LAR, including the cover letter and five enclosures.

My Power
& Light

May 1, 2006

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: My-Plant Units 1 & 2
Docket Nos. 50-001 & 50-002
Administrative Control of Containment Penetrations during Refueling

Pursuant to 10 CFR 50.90, My Power & Light hereby requests the following amendment:

Revise Units 1 and 2 Technical Specifications (TS) for Limiting Condition for Operation (LCO) 3.9.4, "Containment Penetrations," to allow containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere to be open during refueling activities if appropriate administrative controls are established. The proposed changes are consistent with NRC-approved Technical Specification Task Force (TSTF) Traveler TSTF-312, Revision 1 (Reference 1).

The TS and Bases changes, our technical and regulatory evaluation of the changes, and one formal commitment are enclosed.

My Power & Light requests approval of the proposed amendment by August 1, 2007 to support the fall 2007 refueling outage for Unit 1. Once approved, the amendment shall be implemented within 60 days.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 31, 2006.

If you have any questions or require additional information, please contact Mr. I. M. Licensing at 000-111-2222 or iml@mpl.com.

Sincerely,

I. R. Boss
Vice President

Enclosures:
1. Licensee Evaluation
   - Attachment 1 - TS Page Markups
   - Attachment 2 – Changes to TS Bases
   - Attachment 3 – Retyped TS Pages
2. List of Regulatory Commitments

cc: Region 0
I. X. Reviewer, NRR
I. Y. Inspector, Region 0
I. Z. Local, State Contact

## LICENSEE EVALUATION

Subject: Application for Amendment to TS 3.9.4, "Containment Penetrations," to Allow Open Penetrations during Refueling Operations if Appropriate Administrative Controls are Established

## 1.0 SUMMARY DESCRIPTION

This evaluation supports a request to revise OL 50-998 and 50-999 for My Plant Units 1 & 2 to allow reactor containment building penetrations that provide direct access from the containment atmosphere to the outside atmosphere to be open during refueling activities if appropriate administrative controls are established. The penetrations in question are the equipment hatch, the personnel airlock, the emergency airlock, and system penetrations. Currently, TS 3.9.4 requires that containment penetration be closed during core alterations or movement of irradiated fuel inside containment in Modes 5 (cold shutdown) or 6 (refueling) to mitigate the consequences of a fuel handling accident (FHA) during.

The proposed change would revise TS Limiting Condition for Operation (LCO) 3.9.4, "Containment Penetrations," and is consistent with NRC-approved TSTF-312, Revision 1 (Reference 1). A revised FHA for My Plant Units 1 & 2 shows acceptable dose consequences.

Revising TS 3.9.4 to permit open penetrations during core alterations or fuel movement has the following benefits:

- Easier access to and from containment for equipment, personnel, laundry, and trash.
- Faster personnel evacuation from containment in the event of a FHA.
- Easier delivery of equipment to critical path activities inside containment.
- More flexibility in scheduling activities not on the critical path.
- Increased reliability of hatch doors due to reduced wear.
- Reduced traffic through the personnel airlock.
- Cleaner working environment.
- Fewer situations requiring a fire watch.
- Reduced occupational exposure.

In summary, the dose consequences of a FHA inside containment with the containment equipment hatch, airlocks, and other specified penetrations open for the during of the accident release are well within the radiological dose guidelines of 10 CFR 100. We request that NRC approve the proposed amendment based on the operational benefits, additional administrative controls, and acceptable dose consequences.

## 2.0 DETAILED DESCRIPTION

TS 3.9.4 currently precludes opening containment penetrations during operations involving core alterations or fuel movement inside containment. Penetrations that provide direct access from the containment atmosphere to the outside atmosphere must be (1) closed by an automatic isolation valve, a manual isolation valve, a blind flange, or equivalent, or (2) capable of being closed by an operable containment purge and isolation system. Plant procedures establish specific closure controls for containment penetrations.

The proposed change would allow any containment penetration flow path that provides direct access from the containment atmosphere to the outside atmosphere to be open during operations involving core alterations or fuel movement inside containment if appropriate administrative controls are established and maintained. Specifically, the proposed change revises TS 3.9.4(c) by adding a NOTE to permit un-isolating containment penetration flow path(s) under administrative controls during operations involving core alterations or fuel movement inside containment.

The proposed change is consistent with Revision 1 of TSTF-312, "Administratively Control Containment Penetrations." TSTF 312 was approved based on (1) acceptable radiological consequences from a FHA, and (2) the implementation of administrative procedures to ensure that open containment penetrations can and will be promptly closed in the event of a FHA. The My-Plant Units 1 and 2 dose calculations document the time to close the penetrations.

The containment is a barrier to the release of fission products that breach the fuel cladding and reactor coolant pressure boundary during a core-damaging accident. The containment barrier, including penetrations, is designed to limit the release of fission products such that offsite radiation exposure is well below the limits of 10 CFR 100.

During Modes 5 and 6, plant procedures require the capability to close containment within one hour of the loss of shutdown cooling. The closure scope includes the equipment hatch, the personnel airlock, the emergency airlock, and electrical and piping penetrations. Closure controls include guidance to personnel assigned containment closure duties, a list of equipment and materials that must be maintained to assist with containment closure activities, and a list of ongoing activities that affect the capability to close a containment penetration. Penetrations that provide direct access from the containment atmosphere to the outside atmosphere must be (1) closed by an automatic isolation valve, a manual isolation valve, a blind flange, or equivalent, or (2) capable of being closed by an operable containment purge and isolation system. Plant procedures establish specific closure controls for containment penetrations.

### 2.1 Equipment Hatch

The door to the equipment hatch is a welded steel assembly bolted to a double-gasket flange. The hatch is 14 feet in diameter and provides a means for moving large equipment and components into and out of containment during refueling outages. Currently, TS 3.9.4 requires that the door be closed and secured by a minimum of four (of 16) bolts during core alterations or movement of irradiated fuel inside containment.

In 1989, a closure test was conducted as part of an initiative to address Generic Letter 88-17, "Loss of Decay Heat Removal." The test simulated conditions normally found during an outage. The total time required to close the equipment hatch was less than 15 minutes.

Plant procedures will be revised to require a capability for prompt closure whenever the equipment hatch door is open during core alterations or fuel movement inside containment. A designated individual will be assigned to monitor the door to ensure that items that could obstruct closure of the door can quickly be disconnected or otherwise removed. The evaluation described in Section 3 assumes a maximum closure time of 30 minutes.

## 2.2 Airlocks

Personnel transit between the containment interior and the Auxiliary Building through a personnel airlock. Personnel can exit the containment to the outside atmosphere through a smaller emergency airlock. There is a pressure-seating door at each end of each airlock. Currently, TS 3.9.4 requires a minimum of one closed door in each airlock during core alterations or movement of irradiated fuel inside containment.

## 2.3 Other Penetrations

Various plant systems and vent/drain piping have containment penetrations equipped with isolation valves. These penetrations are subject to periodic testing in accordance with the Local Leak Rate Testing (LLRT) program. Currently, TS 3.9.4 requires that these penetrations be closed during core alterations or movement of irradiated fuel within the containment. Therefore, the approximately 40% of containment penetrations that are subject to Type C testing cannot be tested during fuel movement because Type C testing requires an open drain line. The proposed change removes this restriction and significantly improves the logistics for implementing the LLRT program by permitting open penetrations during fuel movement if they can be isolated quickly by an automatic isolation valve, a manual valve, or a blind flange.

## 2.4 Fuel Handling Accident

The FHA analysis in the Updated Final Safety Analysis Report (UFSAR) assumes that a single irradiated fuel assembly (or other heavy load) is dropped onto other irradiated fuel assemblies. The FHA acceptance criteria in Standard Review Plan (SRP) Section 15.7.4 specify that the resulting offsite radiation exposure must be well within the limits of 10 CFRI100. The standard interpretation of "well within" is no greater than 25% of the 10 CFR 100 limits, which translates to 75 rem to the thyroid and 6.25 rem to the whole body.

My Power & Light has reanalyzed the FHA in support of this LAR. The new analysis is described in detail in Section 3 (Technical Evaluation). It assumes that containment penetrations are initially open, and that the limiting pathway (equipment hatch) can be closed within 30 minutes. The resulting offsite exposures remain less than 25% of 10 CFR 100 limits.

## 3.0 TECHNICAL EVALUATION

Compliance with TS 3.9.4 (Containment Penetrations) ensures that the consequences of a postulated FHA inside containment during core alterations or fuel handling remain within acceptable limits. The TS Limiting Condition for Operation (LCO) requires that at least one integral barrier to the release of radioactive material be operable at all times. LCO 3.9.4 requires a closed and bolted equipment hatch, a minimum of one closed door in each airlock, and flanged or valved containment penetrations. Penetrations with automatic isolation valves must be capable of being closed by an isolation signal. As discussed in the TS Bases, isolation methods must be approved and may include the use of temporary barriers during fuel movement.

The changes proposed by this license amendment request are consistent with TSTF-312, Revision 1. They are also consistent with administrative controls in My-Plant Units 1 & 2 TS that permit penetration flow paths to be un-isolated under administrative controls in MODES 1 through 4. The controls include continuous communication between the Control Room and an individual who can isolate the flow path in the event of an accident. Modes 1 through 4 are more limiting than Mode 6 (refueling operations) due to greater stored energy in the RCS and the greater motive force available to disperse radionuclides following a design basis accident.

Similar controls are acceptable for penetrations that are open during core alterations or fuel movement inside containment because the potential for a FHA resulting in containment pressurization is negligible when the reactor is shutdown. Therefore, un-isolated flow path(s) that establish direct access between the containment atmosphere and the outside atmosphere during refueling operations are acceptable provided appropriate administrative controls are in place. The proposed controls include operator awareness of the open penetration and the designation of one or more individuals capable of closing open penetrations in the event of a FHA inside containment.

The My-Plant Units 1 & 2 design basis FHA described in the UFSAR is assumes that a single irradiated fuel assembly is dropped in either the fuel building or the containment. The analyses assume the rupture of the cladding on all fuel rods in the dropped assembly. Conservative assumptions are postulated for safety system design purposes even though administrative controls and physical limitations are imposed during fuel handling operations. Section 15.7.4 of the UFSAR (Reference 3) discusses the consequences of a postulated FHA inside containment. The results from the current FHA analysis indicate an exclusion area boundary thyroid dose of 64.1 REM and a whole body dose of 0.177 REM. These results are well within the 10 CFR 100

(Reference 4) offsite dose limits of 300 REM and 25 REM, respectively, and they are less than the guideline values of Standard Review Plan, Section 15.7.4, Revision 1 (Reference 5).

The limiting event is the FHA inside containment with the personnel airlock doors remaining open. For this event, radionuclides are unlikely to reach the outside atmosphere because there is no pressure differential to drive the dispersion of radioactive material. Administrative controls for prompt closure of the containment penetration flow paths minimize the potential for spreading radioactive isotopes from the containment to the outside atmosphere. Therefore, following a FHA inside containment, the lack of containment pressurization allows sufficient time to manually isolate the penetration flow paths to minimize dose consequences. The consequences of a FHA inside containment with open penetration flow paths are bounded by the current analysis described in the UFSAR. This ensures that the postulated offsite dose is well below 10 CFR 100 regulatory limits and less than the guideline values in Standard Review Plan, Section 15.7.4, Revision 1.

Amendment No. 95 (Reference 6) approved leaving the containment air lock open during fuel movement and core alteration. In that application, My Power & Light recalculated the doses and revised the design basis for the FHA analysis to be consistent with Regulatory Guide 1.25 (Reference 7). In that re-analysis, credit was not taken for the containment building barriers. The analysis calculated the doses at the exclusion area boundary during the first two hours of the event. The calculated doses were within the Standard Review Plan criteria of 6.25 REM to the whole body and 75 REM to the thyroid. As discussed in Amendment No. 107 (Reference 2), the potential dose consequences from a simultaneous release of gaseous effluents through either an un-isolated penetration flow path or open personnel airlock doors is the same. That is because the analysis assumes that all radioactive material from the FHA is released to the environment within a two-hour period. Therefore, allowing penetration flow paths to be un-isolated during core alterations or movement of irradiated fuel does not invalidate the conclusion that the potential dose consequences from a FHA are well below 10 CFR 100 limits.

Historically, the NRC has required containment closure during core alterations and fuel handling as a defense-in-depth measure to limit releases. However, this has been relaxed on a case-by-case basis to permit both personnel airlock doors or selected containment penetrations to be open during core alterations and fuel handling if controls are in place to quickly close one door or isolate the penetration (References 2 and 6). These procedural controls include:

1. Appropriate personnel will maintain an awareness of the open status of the penetration flow path during core alterations and movement of irradiated fuel assemblies within containment.

2. Specified individuals will be designated and readily available to promptly

isolate open penetration flow paths in the event of a FHA inside containment.

Based on the analysis of the FHA and the administrative controls specified for the proposed allowance to un-isolate containment penetration flow paths, the proposed changes are acceptable. With respect to the proposed administrative controls, the proposed license amendment provides assurance that offsite dose levels associated with a FHA inside containment will be maintained well within applicable regulatory limits.

## 4.0 REGULATORY EVALUATION

### 4.1 Significant Hazards Consideration

My Power & Light has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR50.92(c) as discussed below:

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

   Response: No.

   The proposed change would allow the containment equipment hatch door, personnel airlock doors, emergency airlock doors, and other specified penetrations to remain open during fuel movement and core alterations. These penetrations are normally closed during this time period in order to prevent the release of radioactive material in the event of a fuel handling accident (FHA) inside containment. These penetrations are not initiators of any accident. The probability of a FHA is unaffected by the operational status of these penetrations.

   The new FHA analysis with an open containment demonstrates that maximum offsite dose is well within the acceptance limits specified in SRP 15.7.4. The FHA analysis results in maximum offsite doses of 51 rem to the thyroid and 0.18 rem to the whole body. The calculated control room dose is also well within the acceptance criteria specified in GDC 19. The analysis results in thyroid and whole body dose to the control room operator of 0.93 rem and 0.02 rem, respectively

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

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

   Response: No.

   The proposed change does not involve the addition or modification of any plant equipment. Also, the proposed change will not alter the design, configuration, or method of operation of the plant beyond the standard functional capabilities of the equipment. The proposed change involves a TS change that will allow the equipment hatch door, the airlock doors, and other selected penetrations to be open during core alterations and fuel movement inside containment. Open doors and penetrations do not create the possibility of a new accident. Administrative controls will be implemented to ensure the capability to close the containment in the event of a FHA.

   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 amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change has the potential to increase the post-FHA dose at the site boundary and in the control room. However, a revised FHA analysis demonstrates that the dose consequences at both locations remains within regulatory acceptance limits and the margin of safety as defined by Revision 1 of SRP 15.7.4 has not been significantly reduced. To ensure a bounding calculation, the revised FHA was performed with conservative assumptions, for example, it assumes the instantaneous release to the outside atmosphere of all airborne activity reaching the containment. Additional margin will be established through administrative procedures to require that the equipment hatch and at least one door in each airlock be closed following an evacuation of containment.

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

Based on the above, My Power & Light concludes that the proposed amendment does not involve a 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.

## 4.2 Applicable Regulatory Requirements/Criteria

The following table lists the regulatory requirements and plant-specific design bases related to the proposed change.

| TS 3.9.4 |
|---|
| Regulatory Requirements<br><br>• The regulatory basis for TS 3.9.4, "Containment Penetrations," is to ensure that the primary containment is capable of containing fission product radioactivity that may be released from the reactor core following a FHA inside containment. This ensures that offsite radiation exposures are maintained well within the requirements of 10 CFR 100.<br><br>• 10 CFR Part 50, Appendix A (Reference 8), General Design Criterion (GDC) 16, "Design," requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.<br><br>• GDC 19, "Control Room," requires that adequate radiation protection shall be provided to permit access and occupancy under accident conditions without personnel receiving radiation exposure in excess of 5 REM whole body, or its equivalent to any part of the body for the duration of the accident."<br><br>• GDC 54, 'Piping Systems Penetrating Containment," requires that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.<br><br>• GDC 56, "Primary Containment Isolation," describes the isolation provisions that must be provided for lines that connect directly to the containment atmosphere and which penetrate primary reactor containment unless it can be demonstrated that the isolation provisions for a specific class of lines are acceptable on some other defined basis. |

- GDC 61, "Fuel Storage and Handling and Radioactivity Control," requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions.
- The parameters of concern and the acceptance criteria applied are based on the requirements of 10 CFR 100 with respect to the calculated radiological consequences of a FHA and GDC 61 with respect to appropriate containment, confinement, and filtering systems.

Regulatory Guidance:

- UFSAR Section 15.7.4 – The My-Plant Units 1 & 2 design basis Fuel Handling Accident (FHA) is defined as the dropping of a spent fuel assembly onto the spent fuel pool fuel storage area or inside containment. Both analyses assume the rupture of the cladding of all the fuel rods in the assembly. Section 15.7.4 of the UFSAR discusses the consequences of a postulated FHA inside containment.
- 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," describes the methodology used by My-Plant Units 1 & 2 to evaluate the potential radiological consequences of a FHA.
- My-Plant Units 1 & 2 uses NUREG-0800 (Reference 9), U.S. NRC Standard Review Plan, Section 15.7.4, to evaluate system design features and plant procedures provided for the mitigation of the radiological consequences of postulated fuel handling accidents.

Regulatory criteria and guidance are contained in Regulatory Guide 1.25, Section 15.7.4 of NUREG-0800, and NUREG/CR-5009 (Reference 10). The calculated doses are within the Standard Review Plan criteria of 6.25 REM to the whole body and 75 REM to the thyroid.

Section 15.7.4 of the My-Plant Units 1 & 2 UFSAR describes system design features and plant procedures for mitigating the radiological consequences of postulated FHAs. It assumes no credit for iodine removal by the atmosphere filtration system filters. All radioactivity released to the containment is assumed to be released to the environment at ground level over a two-hour period.

### 4.3 Precedent

This request is similar to license amendments issued to Good Power & Light on January 1, 1995, and January 1, 2000, for Good Units 1 & 2. The 1995 amendments permitted the personal and emergency airlocks to be open during core alterations, subject to administrative controls. The 2000 amendments permitted the equipment hatch to be open during core alterations or movement of irradiated fuel, subject to administrative controls. *[NOTE: This section should include a point-by-point comparison between the current LAR and the proposed precedent. All differences should be described and dispositioned as acceptable or not applicable.]*

### 4.4 Conclusions

The technical analysis performed by My Power & Light demonstrates that the dose consequences at the exclusion area and low population zone boundaries are well within the limits of 10 CFR 100. Therefore, the proposed License amendment is in compliance with the General Design Criteria (16, 19, 54, 56, and 61), Regulatory Guide 1.25, NUREG/CR-5009, and Section 15.7.4 of the SRP (NUREG-0800).

## 5.0 ENVIRONMENTAL CONSIDERATION

My Power & Light has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20 (Reference 11), or would change an inspection or surveillance requirement. My Power & Light has evaluated the proposed change and has determined that the change does not involve, (i) a significant hazards consideration, (ii) a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. As discussed above, the proposed changes do not involve a significant hazards consideration and the analysis demonstrates that the consequences from a FHA are well within the 10 CFR 100 limits. The implementation of administrative controls precludes a significant increase in occupational radiation exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51 (Reference 12), specifically 10 CFR 51.22(c)(9). Therefore, pursuant 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

## 6.0 REFERENCES

1. Technical Specification Task Force, TSTF-312, Revision 1, "Administratively Control Containment Penetrations," July 17, 1999.

2. My Power & Light, Operating License Amendment 107, [month day, year].

3. UFSAR Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents."

4. 10 CFR 100, "Reactor Site Criteria."

5. US NRC, Standard Review Plan, Section 15.7.4, Revision 1, "Radiological Consequences of Fuel Handling Accidents."

6. My Power & Light, Operating License Amendment 95, [month day, year].

7. US NRC, 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 (Safety Guide 25)," March 1972.

8. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

9. US NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

10. US NRC, NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors."

11. 10CFR20, "Standards for Protection Against Radiation."

12. 10 CFR 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions."

# TS PAGE MARKUPS

1. Add TS Insert 1 to Page 3.9-6

TS Insert 1

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

# 3.9 REFUELING OPERATIONS

## 3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in each air lock closed; and

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

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

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

**TS Insert 1**

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 |
| | <u>AND</u> | |
| | A.2 Suspend movement of irradiated fuel assemblies within containment. | Immediately |

## CHANGES TO TS BASES

1. Add Bases Insert 1 to Page B3.9-13
2. Add Bases Insert 2 to Page B3.9-14

<u>Bases Insert 1</u>

--- REVIEWERS NOTE ---

The allowance to have containment personnel airlock doors open and penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement and CORE ALTERATIONS is based on (1) confirmatory dose calculations of a fuel handling accident as approved by the NRC staff which indicate acceptable radiological consequences and (2) commitment from the licensee to implement acceptable administrative procedures that ensure in the event of a refueling accident (even though the containment fission product control function is not required to meet acceptable dose consequences) that the open airlock can and will be promptly closed following containment evacuation and that the open penetrations(s) can and will be promptly closed. The time to close such penetrations or combination of penetrations shall be included in the confirmatory dose calculations.

<u>Bases Insert 2</u>

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. Administrative controls ensure that (1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and (2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

| | |
|---|---|
| BACKGROUND (continued) | 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 and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements (Ref.1). |
| 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 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure 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 or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).<br><br>Containment penetrations satisfy Criterion 3 of the NRC Policy Statement. |
| LCO<br><br>**Bases Insert 1** | 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 and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve (continued) |

| | |
|---|---|
| LCO (continued) | closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit. |
| **Bases Insert 2** | |
| 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 potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status. |
| ACTIONS | <u>A.1 and A.2</u><br><br>If the containment equipment hatch, 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 and Exhaust Isolation System 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 | <u>SR 3.9.4.1</u><br><br>This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust vales will demonstrate that the valves are not blocked from closing. Also the Surveillance will |

(continued)

## RETYPED TECHNICAL SPECIFICATION PAGES

Page 3.9-6

Page B 3.9-12

Page B 3.9-12a

Page B 3.9-13

## 3.9 REFUELING OPERATIONS

### 3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in each air lock closed; and

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

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

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

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

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.<br><br><u>AND</u><br><br>A.2 Suspend movement of irradiated fuel assemblies within containment. | Immediately<br><br><br><br>Immediately |

BACKGROUND (continued)

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 and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements (Ref.1).

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 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure 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 or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

LCO

--- REVIEWERS NOTE ---

The allowance to have containment personnel airlock doors open and penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated during fuel movement and CORE ALTERATIONS is based on (1) confirmatory dose calculations of a fuel handling accident as approved by the NRC staff which indicate acceptable radiological consequences and (2) commitment from the licensee to implement acceptable

(continued)

LCO
(continued)

administrative procedures that ensure in the event of a refueling accident (even though the containment fission product control function is not required to meet acceptable dose consequences) that the open airlock can and will be promptly closed following containment evacuation and that the open penetrations(s) can and will be promptly closed. The time to close such penetrations or combination of penetrations shall be included in the confirmatory dose calculations.

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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 and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

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. Administrative controls ensure that (1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and (2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

| | |
|---|---|
| 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 potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status. |
| ACTIONS | <u>A.1 and A.2</u><br><br>If the containment equipment hatch, 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 and Exhaust Isolation System 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 | <u>SR 3.9.4.1</u><br><br>This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust vales will demonstrate that the valves are not blocked from closing. Also the Surveillance will |

(continued)

## LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by My Power & Light in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. I. M. Licensing.

| REGULATORY COMMITMENTS |
|---|
| 1. Include in plant procedures a requirement to close the containment equipment hatch within 30 minutes of a determination that containment must be evacuated. |

## RAI CATEGORIES

The following framework classifies the reasons for RAIs into four categories. The framework is provided for voluntary use by NRC staff and licensees during the RAI process. Licensees may find the framework useful when responding to RAIs, or as a source of lessons learned for improving LAR quality.

1. CATEGORY 1 – the reviewer is requesting additional information because:
    a. the LAR is complex, e.g., power uprate),
    b. the LAR is first-of-a-kind, e.g., based on new technology,
    c. the LAR is affected by an NRC management decision to change regulatory policy,
    d. the LAR proposes the use of new methods/guidance,
    e. the LAR proposes a reduction in safety margin, or
    f. the reviewer has concerns with respect to previously approved methods/guidance.

    Category 1 RAIs are a necessary and expected part of the LAR process.

2. CATEGORY 2 – the reviewer is requesting additional information to evaluate:
    a. input variables or assumptions,
    b. the methodology used or the results obtained,
    c. the applicability or bounding nature of third-party analyses or data correlations,
    d. the differences between the LAR and relevant NRC guidance documents, e.g., Standard Review Plan (SRP), Regulatory Guides, etc.,
    e. the licensees determination that the proposed amendment does not involve a significant hazards consideration,
    f. environmental considerations,
    g. conformance with applicable regulatory requirements,
    h. potentially incorrect information, or
    i. potentially inadequate responses to previous RAIs.

    Category 2 questions highlight the types of information NRC expects to see in a LAR. Thus, licensees can use categorization information to adjust the standard content of LARs to better meet NRC expectations.

3. CATEGORY 3 – the RAI has marginal value because it:
   a. is not directly related to the LAR,
   b. is inconsistent with applicable codes, standards, Regulatory Guides, or SRP sections,
   c. requests information that does not appear to be needed given the precedent cases discussed in the LAR,
   d. requests information that is not safety significant or is not pertinent to the regulatory finding that needs to be made,
   e. requests information that should be known to engineers that work in the general technical area addressed by the LAR,
   f. requests information outside the scope of the current licensing basis (CLB),
   g. requests a formal commitment as a condition of NRC approval,
   h. requests information that is already in the LAR, or
   i. requests information that is accessible from readily available sources that were explicitly referenced in the LAR.

   Designating an RAI as Category 3 does not necessarily mean that it should not be processed as an RAI. For example, a licensee may choose to answer questions that fall in Categories 3.h and 3.i to assist the reviewer in locating pertinent information.

CATEGORY 4 – the RAI does not fall into one of the other three categories.

# PLANT-SPECIFIC ADOPTION OF TSTF TRAVELERS

## D.1 Introduction

The Technical Specification Task Force (TSTF), in consultation with the PWR and BWR Owners Groups, develops generic changes to the Improved Standard Technical Specifications (ISTS). The changes are called "Travelers." If a Traveler is technically acceptable and cost-beneficial, it is submitted to the NRC for review. After a Traveler is approved by NRC, it is given an "A" postscript (e.g., TSTF-445-A) and posted on the TSTF web site (http://www.excelservices.com). Some Travelers are made available for plant-specific adoption in accordance with the Consolidated Line Item Improvement Process (Reference 22). NRR draft Office Instruction LIC-TSTF (Reference 33) describes the overall NRC process for review and approval of Travelers. This includes the budgeting and scheduling of NRC resources for Traveler reviews, the coordination of NRC technical staff review and concurrence, the drafting of a Model Safety Evaluation for each Traveler, and the posting of review status on a public web site.

Reference 20 describes the overall NRC process for managing LAR reviews, including LARs based on Travelers. The adoption of TSTF Travelers promotes consistency among plant-specific Technical Specifications. Thus far, the traveler process has led to several hundred approved changes to the ISTS, most of which have been adopted by individual licensees by means of plant-specific LARs. The process utilizes a standardized format, content, and level of detail that has the following potential advantages:

- avoidance of LARs that are overly detailed
- lower preparation and review costs
- easier comparison of a plant-specific LAR with the generic NRC safety evaluation of the Traveler
- fewer RAIs
- shorter NRC review time

## D.2 Traveler Options

The options for plant-specific adoption of Travelers are:

- adoption of a single Traveler
- adoption of multiple Travelers
- lead plant submittal of a "T" Traveler

## D.3 Adoption of a Single Traveler

Format single-traveler LARs consistent with Appendix A. Provide a level of detail consistent with the following points:

1. Minimize the number and extent of differences between the LAR and the NRC-approved Traveler. If there are differences, they must be fully explained and justified to facilitate NRC review.

2. Maximize the use of cross-references to previously published information presented in the Traveler and NRC approval documentation to minimize the repetition of information. Repetition can be confusing because the NRC reviewer must compare the information restated in the LAR with the information in the Traveler and NRC documentation to ensure there are no differences.

3. The NRC began preparing Safety Evaluations for approved Travelers beginning with TSTF-400. These Travelers can be referenced by number alone.

4. The NRC typically did not prepare Safety Evaluations for approved Travelers numbered less than 400. For most of these Travelers, the NRC provided a letter stating the Traveler was approved, but some of them were approved during public meetings without a letter being written. LARs that reference TSTFs below TSTF-400 should provide the NRC approval date and, if available, an example of a representative plant that has adopted the Traveler, including the approval date and amendment number. The LAR should also discuss any significant differences from the referenced plant-specific LAR. The TSTF maintains a list of plant-specific approvals for these Travelers.

5. In general, it is not necessary to restate the justification for an NRC-approved Traveler. The exceptions are the older Travelers that have limited NRC approval documentation and for which there may not be any adoption precedent. The first adoption of such a Traveler should provide a justification for the change that supplements and is consistent with the justification provided in the Traveler.

6. The LAR adopting an NRC-approved Traveler should contain the following minimum information:
   - Traveler number, approved revision, and title
   - A brief discussion of the change to the plant-specific TS and its relationship to the Traveler
   - Description of differences between the affected plant-specific TS and the ISTS marked up in the Traveler
   - Description of any differences between the Traveler justification and the plant-specific justification
   - Description of any differences between the relevant plant-specific design and the design assumed in the ISTS model plant
   - Detailed description of all commitments

## D.4 Adoption of Multiple Travelers

Format multiple-traveler LARs consistent with the format of a LAR adopting a single Traveler. Provide a level of detail consistent with the following points:

1. The guidance on the level of detail for single-traveler adoptions applies to multiple adoption travelers.

2. Discuss all referenced Travelers in an Appendix to the LAR. Begin the discussion of each Traveler on a new page.

3. The LAR may provide markups of plant-specific TS pages on a Traveler-by-Traveler basis, or it may provide all markups in a single location. A single location is preferred if more than one of the referenced Travelers affect the same TS page(s). If the pages are in a single location, each change should be annotated in the right-hand margin with the corresponding Traveler number.

4. The LAR may provide a separate "no significant hazards consideration" (NSHC) determination for each referenced Traveler, a single NSHC for all referenced Travelers , or multiple generic NSHCs for each separate type of change (i.e., administrative, less restrictive, more restrictive, or relocation). This is similar to the format used for ISTS Conversions [Reference 34]. The approach selected should depend on the number of Travelers being adopted and the complexity of the proposed changes.

5. A LAR that proposes to adopt a large number of Travelers may use an approach similar to an ISTS conversion. During conversions, each change is identified as "administrative," "less-restrictive," "more-restrictive," or "relocation." A "discussion-of-change" section is written for each change. If a TS markup is the same as the ISTS, the discussion-of-change section may reference the relevant Traveler. If not, a more detailed discussion is necessary. A single NSHC is written for administrative, more-restrictive, and relocated items. Individual NSHCs are written for each type of less-restrictive change. Additional guidance is contained in Reference 34.

## D.5 Lead Plant Approach

Some Travelers, called T-Travelers, were determined to not be sufficiently cost-beneficial to justify Owners Group funding of NRC review fees, and were not submitted to the NRC for review and approval. However, the Travelers were sufficiently cost-beneficial to develop and post to the TSTF web site for use as templates for plant-specific license amendments. The "T" stands for "template," e.g., TSTF-445-T. The industry Traveler review process ensures that T-Travelers meet the same ISTS format and usage rules as Travelers that are submitted for generic approval by NRC.

Licensees that submit LARs based on a T-Traveler are encouraged to volunteer as a "lead plant" to sponsor a generic review by NRC that will result not only in a plant-specific license amendment for the lead plant, but will also convert the T-Traveler to an A-Traveler approved by the NRC. Under the lead plant approach, the NRC's plant-specific safety evaluation (SE) will be sufficiently generic to serve as the approval of the Traveler.

The basic steps in the lead-plant approach are described below:

1. State in the LAR cover letter that it is a lead plant submittal for a T-Traveler that has been approved by the Owners Groups and the TSTF. Cite the Traveler number and title. It is recommended that a copy of the T-Traveler be included as an attachment to the submittal. Highlight and justify all differences between the LAR and the Traveler. Forward a copy of the LAR to the TSTF at tstf@excelservices.com.

2. Licensees should recognize that NRC review fees for a generic lead-plant review will likely exceed the review fees for a corresponding plant-specific review. If a licensee decides to withdraw from the lead-plant process after submittal, the NRC review becomes a plant-specific review only, and the NRC will not review the generic aspects of the Traveler.

3. RAI correspondence that affects the generic nature of the Traveler should be coordinated with the TSTF to ensure that any resulting changes continue to follow the ISTS format and usage guidelines for all applicable plant designs. The TSTF will revise the "T" Traveler as necessary to reflect changes.

4. The NRC SE should state that it constitutes regulatory approval of both the plant-specific request and the generic Traveler. A copy of the NRC approval documentation should be forwarded to the TSTF.

5. The TSTF will change the Traveler from a "T" Traveler to an "A" Traveler. The approved Traveler, the lead-plant submittal, RAI correspondence, the NRC SE, and the NRC cover letter are posted on the TSTF web site.

## ACRONYMS

| | |
|---|---|
| ADAMS | Agency Documents Access and Management System |
| ASME | American Society of Mechanical Engineers |
| BWR | Boiling Water Reactor |
| BWROG | Boiling Water Reactors Owners' Group |
| CFR | Code of Federal Regulations |
| CLB | Current Licensing Basis |
| CLIIP | Consolidated Line Item Improvement Process |
| EPRI | Electric Power Research Institute |
| FHA | Fuel Handling Accident |
| FOAK | First of a Kind |
| FSAR | Final Safety Analysis Report |
| GDC | General Design Criteria |
| GIM | Generic Issue Management |
| ISTS | Improved Standard Technical Specifications |
| INPO | Institute of Nuclear Power Operations |
| LAR | License Amendment Request |
| LATF | Licensing Action Task Force |
| LLRT | Local Leak Rate Test |
| LCO | Limiting Condition for Operation |
| LOCA | Loss of Coolant Accident |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| NRR | Office of Nuclear Reactor Regulation |
| NRUG | National Regional Utility Group |
| NSSS | Nuclear Steam System Supplier |
| OGC | Office of the General Counsel |
| OL | Operating License |

| | |
|---|---|
| PM | Project Manager |
| PWR | Pressurized Water Reactor |
| PWROG | Pressurized Water Reactor Owners Group |
| RAI | Request for Additional Information |
| RCS | Reactor Coolant System |
| RIS | Regulatory Issue Summary |
| RUG | Regional Utility Group |
| SE | Safety Evaluation |
| SOC | Statements of Consideration |
| SRP | Standard Review Plan |
| SSC | Structure, System, or Component |
| STARS | Strategic Teaming and Resource Sharing |
| STS | Standard Technical Specifications |
| TIA | Task Interface Agreement |
| TS | Technical Specification |
| TSTF | Technical Specification Task Force |
| TVA | Tennessee Valley Authority |
| UFSAR | Updated Final Safety Analysis Report |
| USA | Utilities Service Alliance |
| USC | U.S. Code |

# REFERENCES


1 10 CFR 50.59, "Changes, tests and experiments," 64 FR 53613, October 4, 1999, as amended at 66 FR 64738, December 14, 2001.

2 NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," November 2000.

3 NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59," November 2000.

4 10 CFR 50.12, "Specific exemptions," 37 FR 5748, March 21, 1972, as amended at 40 FR 8789, March 3 1975 and 50 FR 50777, December 12, 1985.

5 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," 39 FR 1002, January 4, 1972, as amended at 53 FR 36004, September 16, 1988.

6 10 CFR 50.54(a), "Conditions of licenses," [changes to quality assurance program description].

7 10 CFR 50.54(p), "Conditions of licenses," [changes to security plan, or guard training and qualification plan].

8 10 CFR 50.54(q), "Conditions of licenses" [changes to emergency plans].

9 10 CFR 50.55a, "Codes and standards," 36 FR 11424, June 12, 1971, as amended.

10 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," 56 FR 31324, as amended.

11 NEI 97-04, revision 1, "Design Bases Program Guidelines," February 2001.

12 NRC Regulatory Guide 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases," December 2000.

13 NEI 98-03, revision 1, "Guidelines for Updating Final Safety Analysis Reports," June 1999.

14 NRC Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," September 1999.

## REFERENCES

15 NEI 99-04, "Guidelines for Managing NRC Commitment Changes," July 1999.

16 NRC SECY-00-0045, "Acceptance of NEI 99-04, 'Guidelines for Managing NRC Commitments,'" February 22, 2000.

17 NEI 01-01, revision 1, "Guideline on Licensing Digital Upgrades," March 2002 [co-numbered EPRI TR-102348, revision 1].

18 NRC Regulatory Issue Summary (RIS) 2002-22, "Use of EPRI/NEI Joint Task Force Report, 'Guideline on Licensing Digital Upgrades: EPRI TR-102348, Revision 1, NEI01-01: a Revision of EPRI TR-102348 to Reflect Changes to the 10 CFR 50.59 Rule,'" November 25, 2002.

19 NRR Office Instruction LIC-100, Revision 1, "Control of Licensing Bases for Operating Reactors," January 7, 2004.

20 NRR Office Instruction LIC-101, Revision 3, "License Amendment Review Procedures," February 9, 2004.

21 10 CFR 50.90, "Application for amendment of license or construction permit," 64 FR 53614, October 4, 1999.

22 NRC Regulatory Issue Summary 2000-06, "Consolidated Line Item Improvement Process for Adopting Standard Technical Specifications Changes for Power Reactors," March 20, 2000.

23 NRC NUREG series 1430 through 1434, Revision 3.1, "Standard Technical Specifications," December 2005.

24 NEI White Paper, "Standard Format for License Amendment Requests from Commercial Reactor Licensees," October 22, 2002.

25 10 CFR 54.3(a), "Definitions" [license renewal], 60 FR 22491, May 8, 1995.

26 NRC Regulatory Issue Summary 2005-20, "Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, 'Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability,'" September 26, 2005.

27 10 CFR 50.109, "Backfitting," 53 FR 20610, June 6, 1988, as amended at 54 FR 15398, April 18, 1989.

## REFERENCES


[28] NRR Office Instruction COM-106, Revision 1, "Control of Task Interface Agreements," December 24, 2002.

[29] NRC NUREG/BR-0058, revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," September 2004.

[30] NRR Office Instruction LIC-202, "Procedures for Managing Plant-Specific Backfits and 50.54(f) Information Requests," February 10, 2004.

[31] NRR Office Instruction LIC-400, "Procedures for Controlling the Development of New and Revised Generic Requirements for Power Reactor Licensees," February 12, 2004.

[32] 10 CFR 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," 69 FR 2275, January 14, 2004 and 69 FR 25997, May 11, 2004.

[33] NRR Office Instruction LIC-TSTF, "Changing the Standard Technical Specifications by means of Technical Specifications Task Force (TSTF) Travelers," draft 0.2, March 1, 2006.

[34] NEI 96-06, "Improved Technical Specification Conversion Guidance," August 1996