U.S. NUCLEAR REGULATORY COMMISSION

# STANDARD REVIEW PLAN

## 9.5.1 FIRE PROTECTION PROGRAM

**REVIEW RESPONSIBILITIES**

**Primary** - Organization responsible for the review of fire protection

**Secondary** - None

## I. AREAS OF REVIEW

The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth philosophy, that the Commission's fire protection objectives are satisfied. These objectives are: 1) minimize the potential for fires and explosions to occur; 2) rapidly detect, control, and extinguish fires that do occur; and 3) ensure that fire will not prevent the performance of necessary safe-shutdown functions and will not significantly increase the risk of radioactive releases to the environment. In addition, fire protection systems must be designed such that their failure or inadvertent operation does not adversely impact the ability of the structures, systems and components (SSCs) important to safety to perform their safety functions. The FPP for a nuclear power plant licensed to operate generally consists of the following elements:

- comprehensive identification and analysis of fire and explosion hazards
- organization and staff positions responsible for management and implementation of the FPP
- fire prevention program consisting of administrative policy, procedures, and practices for training of general plant personnel; control of fire hazards; inspection, testing and

maintenance of fire protection systems and features; control of plant design and modification; control of fire system outages and impairments; and FPP quality assurance
- automatic fire detection, alarm, and suppression systems, including fire water supply and distribution systems
- manual suppression capability including portable fire extinguishers, standpipes, hydrants, hose stations, fire department connections, fire brigade organization, training, qualification, equipment, and drills; emergency plans and procedures; and, if applicable, offsite mutual aid capabilities,
- building design for fire protection including layout of fire areas, fire barrier design and qualification testing, interior finish, electrical system design, ventilation system design, drainage systems, and other systems and features for minimizing the threat of fire
- post-fire safe-shutdown analysis and procedures that demonstrate that the plant can achieve and maintain safe shutdown in the event of a fire
- probabilistic risk assessment (PRA) that identifies relative fire risks and vulnerabilities

The specific areas of the FPP to be reviewed will vary depending on the type and scope of the applicant's or licensee's submittal. This Standard Review Plan (SRP) can be applied in the review of the FPP for the following submittals:

- applications for new reactor design certifications
- applications for new reactor combined operating licenses (COLs)
- applications to shut down and decommission a licensed plant
- applications for license renewal
- license amendment requests for power uprates
- licensee requests for exemptions and other license amendments that impact FPPs
- other FPP-related submittals, such as fire PRAs

SRP Section 9.5.1 focuses on deterministic FPPs. This SRP section is not intended to be the primary review guidance document for plants that have adopted a risk-informed, performance-based FPP in accordance with 10 CFR 50.48(c) and National Fire Protection Association (NFPA) Standard NFPA 805. The primary review guidance document for NFPA 805 plants will be developed in the future. In the interim, this SRP will be used as appropriate for applications for plants that adopt a performance-based FPP in accordance with 10 CFR 50.48(c).

Unless specifically noted otherwise, the review guidance in this SRP section is applicable to the FPP for new reactor plants.

The staff reviews the FPP described in the licensee's or applicant's submittal with reference to the Acceptance Criteria in this SRP. Specifically, the staff reviews the following to the extent appropriate for the type and scope of the licensee submittals:

1. <u>FPP administration</u> with respect to fire protection organization; administrative policies; fire prevention controls; applicable administrative, operations, maintenance and emergency procedures; quality assurance; access to and egress from fire areas; fire brigade capability; and emergency response capability.

2. Evaluation of the potential fire hazards for areas containing equipment important to safety throughout the plant, for the effect of postulated fires and explosions relative to maintaining the ability to perform safe-shutdown functions, and for minimizing radioactive releases to the environment.

3. Plant layout, access and egress routes with respect to 1) firefighting and local operator manual actions, 2) facility arrangements, and 3) structural design features that provide separation or insolation of redundant systems important to safety.

4. Selection and design of fire detection, alarm, control and suppression systems on the basis of the fire hazards analysis; of design, testing, qualification, and maintenance of fire barriers, including penetration seals; of use of noncombustible materials; and of design of floor drains, ventilation, emergency lighting and communication systems.

5. The fire protection system piping and instrumentation diagrams (P&IDs), including with respect to redundancy of equipment and with respect to the fire protection design criteria and failure modes and effects analysis, including the potential effects of inadvertent discharge or failure of fire protection systems on SSCs important to safety.

6. On multiple unit sites, fire protection and control provisions during construction, shutdown or decommissioning of the adjacent units, in order to verify that the integrity and operability of the shared fire protection systems are maintained and that fire hazards associated with one unit will not have an adverse affect on the adjacent unit.

7. For operating plants and new design applications, post-fire safe-shutdown analysis, including the list of systems and components needed to provide post-fire safe-shutdown capability; the arrangement of the systems and components within the plant fire areas; the separation between redundant safe-shutdown systems and components; the fire protection for safe-shutdown systems and components; and potential interactions between non-safety systems, fire protection systems, and systems important to safety for potential adverse effects on the safe-shutdown capability. New reactor designs must also meet the Commission's enhanced fire protection criteria as described in Appendix A to this SRP section.

8. FPP for shutdown and decommissioned reactors as part of the overall review of the decommissioning plans and activities under 10 CFR 50.82. The staff reviews the fire hazards analysis, fire protection systems and features, and other measures necessary to protect against the release of radioactive material as a result of fire adversely impacting spent fuel storage or radioactive wastes from plant decommissioning, dismantlement, or demolition. The Office of Nuclear Reactor Regulation (NRR) has review responsibility during the initial stages of decommissioning. The Office of Nuclear Material Safety and Safeguards (NMSS), Division of Waste Management and Environmental Protection, Decommissioning Directorate, oversees the decommissioning program after the fuel has been removed from the plant spent fuel pool, including approval of license termination when decommissioning activities are successfully completed.

9. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC). For design certification and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification." The staff recognizes that the review of ITAAC is performed after review of the rest of this portion of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

10. COL Action Items and Certification Requirements and Restrictions. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

11. Operational Program Description and Implementation. For a COL application, the staff reviews the final safety analysis report (FSAR) Table 13.x to ensure the fire protection program is included. The staff reviews the operational program description and the proposed implementation milestones. Specific to this SRP section are the fire protection program based on the requirements of 10 CFR 50.48.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Fire PRAs are reviewed as part of SRP Section 19.0, Probabilistic Risk Assessment. The organization responsible for review of the PRA may consult the organization responsible for fire protection.

2. Guidance for review of plant features that ensure safe shutdown in the event of an intentional attempt to damage plant SSCs (e.g., terrorist attack) is provided in SRP Section 13.6.

3. For COL reviews of operational programs, the review of the applicant's implementation plan is performed under SRP Section 13.4, "Operational Review."

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

## II. ACCEPTANCE CRITERIA

The applicability of the following requirements and acceptance criteria in the conduct of the review is dependent on the type and scope of the submittal. For operating reactors, power uprates and license renewals, the existing plant licensing basis, and specifically the fire protection license condition, establishes the applicability of the acceptance criteria listed below.

For shut-down and decommissioned reactors, only a portion of the criteria is applicable, and the specific criteria of Regulatory Guide (RG) 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown," should provide the basis for the review. For new applications, the criteria in paragraphs 1-6 below are applicable as modified by other relevant criteria, including the enhanced fire protection criteria of SECY 90-016 and SECY 93-087, as well as the passive plant safe-shutdown criteria of SECY 94-084.

The acceptance criteria included in previous revisions of this SRP section as Branch Technical Position SPLB 9.5-1 have been removed and have been incorporated in Revision 1 of RG 1.189, "Fire Protection for Nuclear Power Plants."

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 50.48, "Fire protection," which requires that operating nuclear power plants have a fire protection plan that satisfies General Design Criterion (GDC) 3 and also provides general requirements regarding the content of the fire protection plan and the applicability of 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

2. 10 CFR 50.48(f) establishes the criteria for a fire protection plan for those plants that have submitted the certifications required for license termination under §50.82(a)(1).

3. 10 CFR Part 50, Appendix A, GDC 3, "Fire Protection," establishes the criteria for the fire and explosion protection of SSCs important to safety. GDC 3 also establishes the criteria for fire detection and firefighting systems and for the use of noncombustible and heat-resistant materials throughout the unit.

4. 10 CFR Part 50, Appendix A, GDC 5, "Sharing of Structures, Systems, and Components," as it applies to shared fire protection systems and potential fire impacts on shared SSCs important to safety.

5. 10 CFR Part 50, Appendix A, GDC 19, "Control Room," as it applies to providing the capability both inside and outside the control room to operate plant systems necessary to achieve and maintain safe-shutdown conditions.

6. 10 CFR Part 50, Appendix A, GDC 23, "Protection System Failure Modes," as it applies to safe-failure states of the protection system when exposed to adverse conditions associated with fire events or inadvertent operation of fire protection systems.

7. 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," which establishes the FPP requirements for nuclear power plants operating prior to January 1, 1979, subject to the provisions in 10 CFR 50.48(b). Appendix R establishes, along with other fire protection requirements,

the requirement to demonstrate that one success path of SSCs necessary to achieve and maintain safe shutdown of the reactor is protected from the effects of fire. The substantive provisions of Appendix R, or portions thereof, may apply to plants licensed to operate after January 1, 1979, to the extent incorporated in or provided for in the fire protection licensing basis for the individual plants.

8. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," which establishes regulatory requirements applicable to new reactors.

9. 10 CFR 52.47(a)(1)(vi), as it relates to ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

10. 10 CFR 52.97(b)(1), as it relates to ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license and the Commission's regulations.

11. 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," which establishes regulatory requirements applicable to spent nuclear fuel and waste storage.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for each review described in Subsection I of this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. RG 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," as it applies to the use of PRA in support of changes to the fire protection licensing basis for nuclear power plants. Appropriate techniques for performing a Fire PRA are presented in NUREG/CR-6850 (EPRI TR-1011989), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities."

2. RG 1.188, Revision 1, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," as it applies to FPP considerations for license renewal such as equipment aging issues. This RG endorses the guidance in Nuclear Energy Institute (NEI) document, NEI 95-10, Revision 6, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule."

3. Proposed RG 1.189, Revision 1, "Fire Protection for Nuclear Power Plants," which provides comprehensive staff positions and guidelines on fire protection for nuclear power plants.

4. RG 1.191, “Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown,” which establishes the fire protection objectives and staff positions for implementing fire protection for those nuclear power plants that have submitted the necessary certifications for license termination under 10 CFR Part 50.82(a).

5. Regulatory Guide 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” as it applies to the FPP of any new reactor COL application submitted in accordance with 10 CFR Part 52.

6. Enhanced fire protection criteria for new reactor designs as documented in SECY 90-016, SECY 93-087, and SECY 94-084. SECY 90-016 established enhanced fire protection criteria for evolutionary light water reactors. SECY 93-087 recommended that the enhanced criteria be extended to include passive reactor designs. SECY 90-016 and SECY 93-087 were approved by the Commission in staff requirements memoranda (SRM). SECY 94-084, in part, establishes criteria defining safe-shutdown conditions for passive light water reactor designs.

7. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the fire protection program are reviewed in accordance with 10 CFR 50.48. The operational program for fire protection should be fully implemented prior to fuel loading.

## III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in subsection II.

For each type of submittal, the staff will conduct the review as follows:

1. New Reactor Applications

a. For applications submitted in accordance with 10 CFR Part 50, the staff reviews the preliminary safety analysis report (PSAR) and the FPP in the final safety analysis report (FSAR). All applicable areas of review listed in Section I should be included in the review for a new reactor application. Reviews that cannot be performed adequately at the PSAR stage due to incomplete development of the FPP should be performed at the FSAR stage of the license application. See Appendix A for additional information.

b. For reviews of design certification and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For design certification applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a design certification, an ESP or other NRC-approved material, applications, and/or reports.

After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

c. For all submittals, the staff verifies that the fire protection program is fully described and that implementation milestones have been identified. The staff verifies that the program and implementation milestones are included in FSAR Table 13.x.

The staff will verify the satisfactory implementation of this program by inspection in accordance with NRC Inspection Manual Chapter IMC-2504, "Construction Inspection Program - Non-ITAAC Inspections."

The staff ensures that the program and associated implementation milestone(s) are included within the license condition on operational programs and implementation.

d. The staff provides any necessary support to the organization reviewing fire PRAs in support of new plant design certification applications and COL applications.

2. License Renewal

The staff reviews applications for license renewal to ensure that fire protection SSCs required for compliance with 10 CFR 50.48 are included within the scope of license renewal in accordance with 10 CFR 54.4(a). For those SSCs identified as being in scope, the staff identifies those components that are subject to an aging management review in accordance with 10 CFR 54.21(a)(1). Appendix B of this SRP provides additional guidance for such a review. The staff provides any necessary support to the primary reviewing office for the review of fire PRAs in support of license amendment requests for plant life extension.

3. Power Uprates

The staff reviews license amendment requests for power uprate to ensure that the changes associated with the power uprate do not adversely affect the ability to achieve and maintain safe shutdown following a fire and that regulatory requirements for fire protection continue to be met. Changes to the plant’s power level must be requested and approved via a license amendment, pursuant to 10 CFR 50.90, 50.91, and 50.92. Appendix D of this SRP provides additional guidance for such a review. The staff provides any necessary support to the primary reviewing office for the review of fire PRAs in support of power uprate license amendment requests.

4. License Termination

The staff reviews the FPP for shutdown and decommissioning operations for those plants that have submitted the necessary certifications required by 10 CFR 50.82(a)(1). RG 1.191 provides additional guidance for review of FPPs for shutdown and decommissioning of nuclear power plants.

5. Fire Protection Program Exemptions - Existing Plants

The staff reviews submitted requests for exemption from regulatory requirements applicable to the FPP in accordance with 10 CFR 50.12. The staff reviews the technical justification for the alternative approach and determines whether an exemption is appropriate under the 10 CFR 50.12 guidelines. RG 1.189 provides detailed criteria and guidelines for review of FPP exemption requests, including general conditions for acceptance.

Where fire modeling or fire probabilistic risk assessment methodologies are used as a basis for an exemption request, the review of the exemption will consider the guidance and acceptance criteria for fire modeling provided in RG 1.189, as well as the guidance provided in this SRP, draft NUREG-1824/EPRI 1011999, “Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications,” and NUREG/CR-6850 (EPRI TR-1011989).

6. Fire Protection Program Exemptions - New Reactor Plants

The staff reviews exemptions to an approved certified design for a new reactor in accordance with Section VIII of the appendix to 10 CFR 52 that is applicable to the specific certified reactor design. Exemptions from Tier 1 information are governed by the requirements of 10 CFR 52.63(b)(1), which references the exemption approval process of 10 CFR 50.12.

7. Fire Protection Program License Amendments - New Reactors

The staff reviews license amendments for modifications to, additions to, or deletions from the terms of a new reactor COL, including the ITAAC, in accordance with 10 CFR 52.97(b)(2).

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

1. New Reactor Design Certifications and Combined Operating License Applications

The staff concludes that the applicant’s FPP design criteria and associated implementation are acceptable and meet the applicable requirements of 10 CFR Part 50 and Part 52, and are consistent with Commission policy contained in SECY 90-016, SECY 93-087, and SECY 94-084 (plants with passive safe-shutdown), as well as other applicable acceptance criteria (staff should specify the applicable criteria depending on the type and scope of review). As described above, the staff finds that the applicant has met the guidelines of the applicable regulatory guides and related industry standards.

The applicant has demonstrated that safe shutdown can be achieved even assuming that all equipment in any one fire area (excluding the control room and reactor containment) will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. The applicant's design has provided an independent alternative shutdown capability that is physically and electrically independent of the control room. The applicant' design provides fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the applicant's design ensures that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to an extent that could adversely affect safe-shutdown capabilities, including operator actions.

The applicant has demonstrated that SSCs important to safety, including SSCs that are shared among multiple units, are adequately protected from the effects of fires and explosions. The applicant's design has used noncombustible and heat resistant materials whenever practical and has provided fire detection, suppression, and firefighting capabilities of appropriate capacity and capability to minimize the adverse effects of fire on SSCs important to safety.

The staff concludes that the proposed ITAAC for the FPP provide reasonable assurance that the implementation of the FPP will be in accordance with the approved design and operational program descriptions (where applicable). The staff has included FPP and its implementation milestones within the license condition on operational program implementation.

The staff concludes that for differences between the licensee's FPP and the SRP acceptance criteria, the proposed alternatives provide an acceptable method of complying with the NRC regulations.

2. <u>License Amendments and Exemption Requests</u>

The staff concludes that the proposed exemption or amendment to the licensee's FPP is acceptable and that the FPP continues to meet the applicable requirements of 10 CFR Part 50, 10 CFR Part 52, 10 CFR Part 54 and the enhanced fire protection requirements (new reactors), as well as other applicable acceptance criteria (staff should specify the applicable criteria depending on the type and scope of review). The staff has reviewed the licensee's analysis and justifications for the changes and concludes that the plant is still able to achieve and maintain safe-shutdown conditions and to mitigate a radiological release following a fire.

3. <u>Shutdown/Decommissioning Fire Protection Programs</u>

The staff concludes that the FPP (or related changes) for shutdown and decommissioning of the plant is acceptable and meets the requirements of 10 CFR 50.48(f) and other applicable acceptance criteria, including the guidance in RG 1.191. In meeting the acceptance criteria, the applicant for license termination has demonstrated that radioactive materials are adequately protected from the effects of fires and that potential radioactive hazards to the public, environment, and plant personnel are minimized.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

## V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of exemption requests, license amendments, license applications, and other FPP-related submittals pursuant to 10 CFR Part 50, 10 CFR Part 52, 10 CFR Part 54 and 10 CFR Part 72, as applicable. 10 CFR 50.34(h) requires that each application for a license docketed after May 17, 1982 should include an evaluation of the facility against the SRP of record, including identification and description of all differences between the design features, analytical techniques, and procedural measures proposed for a facility and those in the SRP acceptance criteria.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.

Implementation schedules for conformance with the requirements or guidance discussed herein, if any, are contained in the referenced regulations, regulatory guides, and generic communications.

## VI. REFERENCES

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
   a. 50.12, "Specific exemptions"
   b. 50.34, "Contents of applications; technical information"
   c. 50.48, "Fire protection"
   d. 50.82, "License termination"
   e. 50.90, "Application for amendment of license or construction permit"
   f. 50.91, "Notice for public comment; State consultation"
   g. 50.92, "Issuance of amendment"

2. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licences for Nuclear Power Plants."
   a. 52.47, "Contents of applications"
   b. 52.63, "Finality of standard design certifications"
   c. 52.79, "Contents of applications; technical information"
   d. 52.83, "Applicability of part 50 provisions"
   e. 52.97, "Issuance of combined licenses"

3. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
   a. 54.4, "Scope"
   b. 54.21, "Contents of application - technical specifications"

4. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
    a. General Design Criterion 3, "Fire Protection"
    b. General Design Criterion 5, "Sharing of Structures, Systems, and Components"
    c. General Design Criterion 19, "Control Room"
    d. General Design Criterion 23, "Protection System Failure Modes"

5. 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

6. 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste."

7. Branch Technical Position (BTP) SPLB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." (Formerly BTP CMEB 9.5-1)

8. APCSB 9.5-1, Appendix A, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976."

9. ANS-58.23-200X, "Standard on Methodology for Fire PRA," American Nuclear Society (draft).

10. RG 1.139, "Guidance for Residual Heat Removal." (for Comment)

11. RG 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis."

12. RG 1.188, Revision 1, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses."

13. RG 1.189, Revision 1, "Fire Protection for Nuclear Power Plants."

14. RG 1.191, "Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown."

15. RG DG-1145, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

16. NUREG-0933, "A Prioritization of Generic Safety Issues."

17. NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

18. NUREG-1801, Revision 1, "Generic Aging Lessons Learned (GALL) Report."

19. NUREG-1824/EPRI 1011999, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." (Draft for Comment)

20. NUREG/CR-6850 (EPRI TR-1011989), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities."

21. SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements.” (ML#003707849)

22. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.” (ML#003707849)

23. SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs.” (ML#003708068)

24. Staff Requirements - SECY 93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, July 21, 1993. (ML003708056)

25. Staff Requirements - SECY 90-016 - Evolutionary Light-Water Reactor (ALWR) Certification Issues and Their Relationship to Current Regulatory Requirements, June 26, 1990. (ML003707885)

26. IN 2002-27, “Recent Fire at Commercial Nuclear Power Plants in the United States.”

27. NEI 95-10, Revision 6, “Industry Guide for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule.”

28. NFPA 804, “Fire Protection for Advanced Light Water Reactors.”

29. NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants.”

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## APPENDIX A
## Supplemental Fire Protection Review Criteria for New Reactors

Unless specifically noted otherwise, the review guidance in this SRP section is applicable to the FPP for new reactor plants. This appendix provides additional guidance applicable to new reactor FPPs.

Many of the current fire protection requirements and guidelines for operating reactors were issued after the construction permits and/or operating licenses were approved by the Commission. The backfit of these requirements and guidelines to existing plant designs created the need for considerable flexibility in the application of the regulations on plant-by-plant basis. For new reactor designs, fire protection requirements, including the protection of safe-shutdown capability and the prevention of radiological release, can be readily integrated in the planning and design phase for the plant.

For applications submitted in accordance with 10 CFR Part 52, design elements of the FPP are addressed in the design certification process. During the design certification process, action items are identified that should be addressed by the combined license applicant. These commitments include action items to establish the FPP for protection of SSCs important to safety as well as the procedures, equipment, and personnel necessary to implement the program. These commitments include, but are not limited to, updating the fire hazards analysis to address final plant design and administrative program elements (e.g., licensee fire protection staffing and organization, quality assurance, procedures, fire prevention programs, and training); fire brigade and emergency response capability; the final design of fire protection systems and features; and the design and analysis of post-fire safe-shutdown capability.

The review of COL applications should also consider the guidance to applicants provided in DG-1145, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

1. Enhanced Fire Protection Criteria

Based on operational experience with existing reactors and insights from examination of internal fire events, the staff determined that fire protection for safe-shutdown capability should be enhanced for new reactor designs. The enhanced fire protection criteria were initially proposed to the Commission in SECY-90-016. This criteria was extended to the review of passive LWR designs in SECY-93-087. These criteria are as follows:

> Evolutionary advanced light water reactor (ALWR) designers must ensure that safe shutdown can be achieved assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. Evolutionary ALWRs must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the evolutionary ALWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions.

2. Passive Plant Safe-Shutdown Condition

As discussed in SECY-94-084, the definitions of safe shutdown as contained in the Commission's regulations and guidelines do not address the inherent limitations of passive residual heat removal systems.

In General Design Criterion (GDC) 34 of Appendix A to 10 CFR Part 50, the NRC regulations require that the design include a residual heat removal (RHR) system to remove residual heat from the reactor core so that specified acceptable fuel design limits are not exceeded. GDC 34 further requires suitable redundancy of the components and features of the RHR system to ensure that the system safety functions can be accomplished, assuming a loss of offsite power or onsite power, coincident with a single failure. The NRC promulgated these requirements to ensure that the RHR system is available for long-term cooling to ensure a safe-shutdown state.

Post-fire safe shutdown for currently operating LWRs is defined in RG 1.189 as those conditions specified in the Technical Specifications for Hot Standby [Pressurized Water Reactors (PWRs)], Hot Shutdown [Boiling Water Reactors (BWRs)], and Cold Shutdown. RG 1.139 specifies Cold Shutdown as 93.3 °C (200 °F) for PWRs and 100 °C (212 °F) for BWRs.

Passive reactor designs are limited by the inherent ability of the passive heat removal processes and cannot reduce the temperature of the reactor coolant system below the boiling point of water for heat transfer to occur between the reactor coolant and the heat sink. The plant designs include cooling systems to bring the reactor to cold shutdown or refueling condition; however, these systems are not safety grade. These non-safety-grade systems (i.e., makeup water to the heat sink and cool-down capability) are necessary to maintain long-term cooling (i.e., beyond 72 hours) and should be capable of accomplishing their respective functions without damage to the fuel as demonstrated by design and analysis.

Based on the discussion and recommendations of SECY-94-084, the passive decay heat removal systems should be capable of achieving and maintaining 215.6 °C (420 °F) or below for non-loss-of-coolant-accidents (non-LOCA) events. This safe-shutdown condition is predicated on demonstration of acceptable passive safety system performance and the acceptable resolution of regulatory treatment of non-safety systems that are necessary for long-term shutdown.

3. Applicable Industry Codes and Standards

In general, the FPP for new light water reactor designs should comply with the provisions specified in NFPA 804, "Fire Protection for Advanced Light Water Reactors," related to the protection of post-fire safe-shutdown capability and the mitigation of a radiological release resulting from a fire. However, the NRC has not formally endorsed NFPA 804, and some of the guidance in the NFPA standard conflicts with regulatory requirements. Where conflicts occur, the applicable regulatory requirements and guidance will govern. The standards of record related to the design and installation of fire protection systems and features sufficient to satisfy NRC requirements in all new reactor designs are those NFPA codes and standards in effect 180 days prior to the submittal of the application under 10 CFR Part 50 or 10 CFR Part 52.

## 4. Other New Reactor Designs

FPPs for proposed new non-light water reactor designs should meet the overall fire protection objectives outlined in RG 1.189 related to safe shutdown and radiological release, as well as the specific fire protection requirements where applicable. Fire hazards should be identified by the applicant, evaluated, and an appropriate level of protection provided to meet these objectives. Design reviews and testing programs should confirm the safe-shutdown capability. SSCs important to safety should be protected in accordance with the enhanced criteria described above for light water reactors. Fire protection systems and features should be consistent with the RG 1.189 criteria to the extent a fire hazards analysis conducted by the applicant shows it to be necessary.

## 5. Fire Protection Program Implementation Schedule

Fire protection has been identified as an "operational program" in SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria." However, only those elements of the FPP that will not be fully implemented until the completion of the plant should be addressed as an operational program. These elements may include, but are not limited to, the fire brigade capability, combustible and ignition source control program, procedures and pre-fire plans, and portable extinguishing equipment. The COL application should identify the operational program aspects of the FPP and the implementation schedule for each. The staff should develop a license condition with respect to the implementation milestones. In lieu of the implementation schedule, the applicant may propose inspections, tests, analyses, and acceptance criteria for these aspects of the program.

## 6. Risk-Informed Review of New Reactors

Review of existing reactors' license applications included an assessment of the plants' FPPs without guidance as to the relative risk significance of one aspect of the program over another. While the current fire protection regulations and guidance are risk-informed to a certain extent, they do not provide a basis for focusing staff resources on the most risk-significant areas of fire protection. The experience gained from regulating and inspecting existing plants has identified aspects of the plant FPPs that warrant more extensive review. In addition, while a risk-informed approach to new reactor design review should reflect the experience gained in connection with existing plants, the new reactors include significant design improvements that impact the FPP. These design improvements should also be considered when reviewing a license application for a new reactor plant. Finally, in addition to the Browns Ferry fire, there have been other notable plant fires that have provided insight with respect to specific nuclear plant fire risks and how to protect against them (see e.g., IN 2002-27, "Recent Fires at Commercial Nuclear Power Plants in the United States"). The following discussion of the relative risk significance of the various aspects of a plant FPP applies to all new reactors, whether or not they adopt a risk-informed, performance-based FPP.

### 6.1 Primary Focus of Staff Review

Since the new reactor approach to protection of post-fire safe-shutdown capability is to provide installed passive separation of redundant trains, the staff review should focus on the licensee's approach to train separation. The staff should review the detailed definition of train separation; the method of identifying which systems, components and circuits need to be separated; the assumptions upon which adequate separation is determined; the design and testing of the separation barriers; the approach when full separation is not feasible; the method of verifying that the separation barrier is installed and maintained properly; and the method of verifying that the as-built cable routing provides the separation necessitated by the design.

6.2 Aspects of New Reactor Fire Protection Programs that Reduce Fire Risk

The overall maturity of fire protection regulation, nuclear plant operation, and analysis methods and the opportunity to incorporate the benefits in the original plant design will greatly enhance new reactor plant safety. The following aspects of the new reactor FPPs will also enhance post-fire plant safety and should be considered by the staff when reviewing license applications:

a. The enhanced fire protection concept and fully-separated 4-train design reduce the safety significance of fire detection/suppression systems, fire brigade response, and other aspects of the FPP for the areas of the plant where the enhanced level of fire protection is provided.

b. Where the plant's design includes an additional safe-shutdown train to ensure safe-shutdown capability when one train is out for maintenance (i.e., there are at least three 100%-capacity redundant trains) and one train fails due to fire, the maintenance downtime for any one train is likely to be a small percentage of total operating time. Consequently, there may be a high probability that even with loss of one train from fire, an extra train beyond the minimum required for safe shutdown will be available.

c. Since the fire protection regulations are being incorporated in the original design rather than being backfitted to existing plants, use of the plant change process should be greatly reduced, which should reduce the potential risk increases due to changes.

d. Post-fire, safe-shutdown circuit analysis should be greatly simplified, reducing the potential for errors.

e. Full train separation should significantly reduce security concerns associated with a fire by reducing access needs.

f. Extensive use of fiber optics should greatly reduce the likelihood of hot shorts and spurious actuations - this development is particularly significant in the control room where full separation of trains is not possible.

g. Use of fiber optics also reduces the fire area combustible loadings and thus the challenge to fire barriers.

h. The enhanced fire protection approach should greatly reduce the importance and scope of previously contentious fire protection issues such as operator manual actions and multiple spurious actuations.

i. The concept of alternative/dedicated shutdown systems, widely used in current reactors, should be virtually eliminated for new reactors (except for a control room or containment fire).

j. Enhanced fire protection attention to smoke migration and smoke damage should reduce the contribution of these phenomena to overall fire risk.

k. The increased level of passive protection necessary for new reactor designs reduces the potential contribution to overall fire risk from delay in applying water to electrical fires.

l. Use of digital control systems greatly reduces the number and size of electrical cabinets in the control room, reducing (likely to a significant extent) the fire ignition frequency in this critical area.

m. Where used, gel-type batteries virtually eliminate the hydrogen gas explosion hazard in plant battery rooms.

n. Reactors with passive shutdown systems have reduced combustible loading, reduced ignition sources, and reduced potential for fire-induced equipment failure.

o. Use of PVC and other non-IEEE 383 rated cable jacketing and insulation should be minimized.

p. The Advanced Boiling Water Reactor (ABWR) and the Economic Simplified Boiling Water Reactor (ESBWR) design plants have no external reactor coolant pumps, eliminating a major fire hazard inside containment. In addition, the containment atmosphere during operation of the ABWR and ESBWR is inerted as with the existing BWR plants.

6.3 Risk-Informed Post-Fire Safe-Shutdown Circuit Analyses

RIS 2004-03, Rev. 1, "Risk-Informed Approach for Post-Fire Safe-Shutdown Circuit Inspection," was issued to provide NRC inspectors with guidance in performing risk-informed inspections of existing plant post-fire safe-shutdown circuit protection. This guidance may also be considered in the review of new reactor circuit analyses; however, it is important to note that the guidance in this RIS is not a basis for regulatory compliance. The guidance optimizes inspector resources by identifying the most risk-significant circuit configurations and cable materials. A plant (including a new reactor) that has not adopted a risk-informed, performance-based FPP may not apply the information included in this RIS to deviate from regulatory requirements without the review and approval of the NRC in accordance with the exemption process (10 CFR 52.63(b)(1) and 10 CFR Part 52.93, as applicable). NUREG/CR-6850 provides guidance with respect to fire PRA that may be applied to a risk-informed post-fire safe-shutdown circuit analysis. The staff may consider the information in RIS 2004-03, Rev. 1 and NUREG/CR-6850 when evaluating exemption and license amendment requests that are based on risk-informed methodologies.

6.4 Additional Risk Consideration for New Reactor Fire Protection Programs

Turbine buildings remain potentially high-fire-risk areas in new reactor plants. Consideration should be given to the potential risk to adjacent safety related buildings and to ensuring control room or remote shutdown station habitability in the event of a major turbine fire.

7. Fire Protection for Non-Power Operation

NRC regulations and guidance do not specifically address fire protection during non-power modes of plant operation (e.g., during shutdown for maintenance and/or refueling) except for existing plants that adopt an NFPA 805 FPP. However, the guidance for fire prevention in Regulatory Position 2 of RG 1.189 is applicable to all modes of plant operation, including shutdown. License applications for new reactors should also address any special provisions to ensure that, in the event of a fire during a non-power mode of operation, the plant can be maintained in a safe and stable condition.

8. Fire Protection System as Backup to Safety-Related Systems

Where a portion of the fire protection system provides required backup to a safe-shutdown system (e.g., makeup to a passive shutdown cooling system), it must meet the design basis requirements and other regulatory requirements of the safe-shutdown system for which it is providing backup. Although the fire protection system must be designed to perform both the required safe-shutdown function and the required fire protection function, the system need not be designed to perform both functions simultaneously.

9. Alternative Designs and Non-Applicable Acceptance Criteria

The new reactor designs that have been reviewed by the NRC have proposed FPP approaches for specific areas of the plant that are not in accordance with the acceptance criteria in RG 1.189. In addition, some of the acceptance criteria in RG 1.189 are not applicable to some reactor designs. The following are examples of alternative designs that have been accepted by the NRC and plant design features for which the acceptance criteria do not apply. These are examples and may not include all cases.

9.1 Alternative Designs

a. At least one new reactor design has been certified by the NRC without meeting the guidance in RG 1.189, Regulatory Position 6.1.2.2, to provide detection in control room cabinets and consoles. The acceptance of this approach was based on the low combustible loading in these cabinets and on the continuous occupancy of the control room, which allows rapid detection and response to a fire in the control room. Acceptance of a similar alternative design for other new reactor designs should be based on the fire hazards analysis.

b. At least one new reactor design has been certified by the NRC without meeting the guidance in RG 1.189, Regulatory Position 6.1.2.1, to provide area automatic fire suppression for control room under-floor areas and ceiling areas. The acceptance of this approach was based on the low combustible loading in these areas and on the continuous occupancy of the control room, which allows rapid detection and response to a fire in the control room. Acceptance of a similar alternative design for other new reactor designs should be based on the fire hazards analysis.

c. At least one new reactor design has been certified by the NRC without meeting the guidance in RG 1.189, Regulatory Position 6.1.2, to provide automatic water suppression in peripheral rooms in the control room complex. The acceptance of this approach was based on the low combustible loading in these areas and on the continuous occupancy of the control room, which allows rapid detection and response to a fire in the control room complex. Acceptance of a similar alternative design for other new reactor designs should be based on the fire hazards analysis.

d. The standpipes and hose stations serving the ESBWR containment are located outside of the containment (the acceptance criteria in RG 1.189, Regulatory Position 6.1.1.2, state that the standpipe and hose stations should be located outside of the drywell). The staff found this arrangement to be acceptable because it provided the capability to reach all areas inside the containment with at least one hose stream. The ESBWR containment is inerted during normal power operation and there are multiple access hatches around the perimeter of the containment. This arrangement may also be acceptable for other new reactor designs with inerted containments if the staff finds access and hose station capability is acceptable.

9.2 <u>Non-Applicable Acceptance Criteria</u>

a. In at least one new reactor design (ESBWR), the standby diesel generators are not required for safe shutdown. If these diesel generators are not important to safety, the guidance in RG 1.189 for diesel generator rooms is not applicable (unless the fire hazards analysis identifies an exposure hazard from the cable spreading room to adjacent areas containing equipment or cables important to safety). The staff should consider the diesel generators' importance to safety, as well as the potential impact on adjacent SSCs, when reviewing the fire protection provisions for these areas.

b. Cable spreading rooms typically include circuits that are important to safety and that, therefore, should be protected from fire in accordance with the acceptance criteria. The cable spreading rooms in at least one new reactor design (ESBWR) do not contain any electrical cables or equipment important to safety. The guidance in RG 1.189 for cable spreading rooms is not applicable to these cable spreading rooms (unless the fire hazards analysis identifies an exposure hazard from the cable spreading room to adjacent areas containing equipment or cables important to safety). The staff should consider the cable spreading rooms' importance to safety, as well as the potential impact on adjacent SSCs, when reviewing the fire protection provisions for these areas.

## APPENDIX B
## Supplemental Fire Protection Review Criteria for License Renewal

The purpose of this appendix is to provide guidance on the review of the fire protection system in an application for renewal of a nuclear power plant operating license submitted in accordance with the provisions of 10 CFR Part 54, “Requirements for Renewal of Operating Licences for Nuclear Power Plants.” RG 1.188, “Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses,” provides additional information and guidelines on the renewal process. The RG endorses the methods contained in NEI guideline, NEI 95-10, “Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule,” Revision 6, June 2005. NUREG-1800, Revision 1, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants” and NUREG-1801, Revision 1, “Generic Aging Lessons Learned (GALL) Report” also provide review guidance for license renewal applications.

10 CFR 54.4(a)(3), states, in part, that SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission’s regulations for fire protection (10 CFR Part 50.48) are within the scope of the rule.

NUREG-1800 and NEI 95-10 provide the methodology for scoping and screening of fire protection SSCs. When evaluating license renewal applications, it is important to note that the scope of SSCs included in 10 CFR Part 50.48 goes beyond the protection of only safety-related equipment. In accordance with General Design Criterion (GDC) 3, “Fire Protection,” the scope of equipment required to comply with 10 CFR Part 50.48 is broad and also includes fire protection SSCs needed to minimize the effects of a fire and to prevent the release of radioactive material to the environment - i.e., equipment “important to safety.” If applicable, the scoping methods used by an applicant should include review of any commitments made for compliance with Appendix A to Branch Technical Position APCSB 9.5-1, “Guidelines for fire Protection for Nuclear Plants Docketed Prior to July 1, 1976,” or 10 CFR Part 50, Appendix R, “Fire Protection Program For Nuclear Power Facilities Operating Prior to January 1, 1979.”

10 CFR Part 54.21 states that for those components with intended functions that are identified within the scope of license renewal, those components which are passive (do not perform their functions with moving parts) and long-lived (are not subject to replacement based on qualified life or routine replacement) are subject to an aging management review (AMR). Examples of fire protection components which are passive and long-lived, and that, therefore, would be subject to an AMR, include fire barrier assemblies (e.g. ceilings, damper housing, doors, floors, penetration seals and walls), sprinkler heads, fire suppression system piping and valve bodies, and fire protection tanks and pump casings, and fire hydrant casings. Active components are defined as components that perform an intended function as described in 10 CFR 54.4 with moving parts or with a change in configuration or properties, and they are excluded from the AMR. For example, smoke/heat detectors are considered active components.

Certain passive and long-lived components are considered consumables and, therefore, are not subject to inclusion in the AMR. System filters, fire extinguishers, fire hoses, and air packs (within the scope of license renewal) may be excluded, on a plant-specific basis, from an AMR under 10 CFR Part 54.21(a)(1)(ii). These components are considered to be within the scope of license renewal and are typically replaced based on specific performance and condition monitoring activities that clearly establish a routine replacement practice based on a qualified life of the component. These components may be excluded from an AMR based on specific performance and condition monitoring activities, provided that the applicant (1) identifies and lists in the license renewal application each component type subject to such replacement, and (2) identifies the applicable monitoring and replacement programs that conform to appropriate standards (e.g., NFPA standards).

The applicant should state in the license renewal application that the components are included within scope but excluded from an AMR on the basis of the consumables position. In addition, the application should identify those fire protection system components that the licensee considers to be outside of the scope of equipment required for 10 CFR 50.48 compliance as well as the basis for that determination. The license renewal application should include an up-to-date piping and instrumentation diagram for the fire protection system that clearly indicates the in-scope portions of the system.

For all components identified within the scope of license renewal and subject to an AMR, programs must be in place to maintain each component's intended function throughout the period of extended operation. NUREG-1801 identifies aging management programs that were determined to be acceptable to manage aging effects of SSCs in the scope of the license renewal as required by 10 CFR 54. For example, the intended function of fire suppression piping or the fire pump casing is to provide a pressure boundary. Programs to manage the aging effects of the pressure boundary can be existing plant programs, modified (or enhanced) programs, or new programs specifically created to address aging concerns. The development of modified or newly created programs is dependent upon (1) the aging effect that needs to be managed, and (2) the ability of the current program to manage the aging effect throughout the period of extended operation.

Plants that have installed Halon 1301 extinguishing systems that will be credited during the extended life of the plant should have either a plan for continued access to an adequate supply of replacement Halon or a plan to replace the system.

Due to the uniqueness of each existing nuclear power plant and to the variations in plant licensing bases, the staff should consider that requirements imposed on one plant are not necessarily applicable to another plant and, similarly, exceptions approved for one plant may not apply to another plant. Each plant should be evaluated based on the site-specific design and licensing basis.

## APPENDIX C
## Supplemental Fire Protection Review Criteria for Fire Probabilistic Risk Assessments (PRA)

The purpose of this appendix is to provide guidance for the review of the fire protection information to be provided in an application for PRA. An existing plant that has not adopted a risk-informed, performance-based FPP in accordance with 10 CFR 50.48(c) may apply risk-informed methodologies, including fire PRA, to the evaluation of a FPP change. However, the proposed methodologies, including the acceptance criteria, must be reviewed and approved by the NRC prior to the implementation of the plant change.

10 CFR Part 52.47(a)(v) requires that new reactor applications submitted under Part 52 include a design specific probabilistic risk assessment. A detailed fire PRA is not necessarily required for a new reactor FPP. However, if a COL applicant references a certified design and if that certified design developed a fire PRA, then the COL applicant, per proposed 10 CFR 52.80(a), is to use that PRA and update it to reflect site and plant-specific information that may not have been available at the design stage. In addition, a licensee that has a risk-informed, performance-based FPP (similar to an NFPA 805 program) or that plans to evaluate plant changes using a risk-informed approach must have a detailed fire PRA.

The term “fire PRA” encompasses all levels and types of PRAs, ranging from a simplified bounding analysis to a detailed analysis in accordance with NUREG/CR-6850 and the draft American Nuclear Society Fire PRA Standard. NUREG/CR-6850 should provide the basis for the review of the proposed methodologies. Refer to SRP Chapter 19, “Probabilistic Risk Assessment,” for additional guidance on the review of nuclear power plant PRAs.

A fire PRA should be subjected to a peer review to the extent that adequate industry guidance is available. The industry guidance will be reviewed and, if appropriate, accepted by the NRC prior to its application to specific fire PRAs. The results of the plant-specific peer reviews should also be reviewed by the NRC. A peer review should be conducted for all types and levels of fire PRAs. In the event that adequate industry guidance is not available for conducting a fire PRA peer review, the NRC should review the fire PRA for acceptability.

Licensees may use PRA and/or risk insights gained from other methods in support of proposed changes to the plant licensing basis, such as license amendment requests pursuant to 10 CFR 50.90 and 50.92. RG 1.174, “An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” provides guidelines for the use of PRA in support of plant changes that require NRC approval. Plant changes that are not subject to NRC approval are not within the scope of RG 1.174. Where PRA is used by licensees in support of submittals to change the plant licensing basis, the guidelines of SRP Chapter 19 should be followed.

Licensees may apply fire modeling methodologies to a performance-based evaluation of the FPP and to changes to the program. Fire modeling results can provide input to a change evaluation, but the change should also be evaluated for the impact on plant risk, defense-in-depth, and safety margin. Licensees should document that the fire models and methods used meet NRC requirements. The licensee should also document that the models and methods used in performance-based analyses are used within their limitations and with the rigor required by the nature and scope of the analyses. These analyses may use simple hand calculations or more complex computer models, depending on the specific conditions of the scenario being evaluated.

The NRC's Office of Nuclear Regulatory Research (RES) and the Electric Power Research Institute (EPRI) have documented the verification and validation (V&V) for parts of five fire models in draft NUREG-1824/EPRI 1011999, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." The specific fire models documented are (1) NUREG-1805, "Fire Dynamics Tools (FDT$^s$)," (2) Fire-Induced Vulnerability Evaluation (FIVE), Revision 1, (3) the National Institute of Standards and Technology (NIST) Consolidated Model of Fire Growth and Smoke Transport (CFAST), (4) the Electricité de France (EdF) MAGIC code, and (5) the NIST Fire Dynamics Simulator (FDS).

Licensees may propose the use of fire models that have not been specifically V&V'd by the NRC; however, licensees are responsible for providing acceptable V&V of these fire models. The V&V documents for licensee-proposed fire models are subject to NRC review and approval.

## APPENDIX D
## Supplemental Fire Protection Review Criteria for Power Uprates

The purpose of this appendix is to provide guidance for the review of the fire protection information in an application for a power uprate. Power uprates typically result in an increase in decay heat generation following a plant trip; however, this change usually does not affect the elements of a FPP related to administrative controls, fire suppression and detection systems, fire barriers, the fire protection responsibilities of plant personnel, the procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown, nor does it usually result in an increase in the potential for a radiological release resulting from a fire. The licensee's submittal should confirm that the power uprate results in no changes to these elements, and this finding should be reflected in the staff's safety evaluation. If the licensee indicates that there is an impact on these elements, the staff should review the impact against the acceptance criteria in the applicable sections of this SRP to ensure that the Commission's fire protection goals are satisfied.

The systems relied upon to achieve and maintain safe shutdown following a fire may be affected by the power uprate due to the increase in decay heat generation following a plant trip. For fire events where the licensee is relying on one full train of the redundant systems normally used for safe shutdown, the licensee's analysis of the impact of the power uprate on the important plant process parameters performed for other plant transients, such as a loss of off-site power or a loss of main feedwater, will typically bound the impact for a fire event such that a specific analysis for fire events is not necessary. However, where a licensee relies on less than full capability systems for fire events, such as partial automatic depressurization or a reduced capability makeup pump, the licensee should provide a specific analysis for fire events that demonstrates that the fuel design limits are not exceeded, that fuel integrity is maintained and that there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Licensees that rely on alternative/dedicated or backup shutdown capability for post-fire safe shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability. The staff should verify that the alternative/dedicated or backup systems relied upon for post-fire safe shutdown are capable of achieving and maintaining safe shutdown considering the impact of the power uprate.

The plant's post-fire safe-shutdown procedures may also be impacted by the power uprate. For example, the allowable time to perform necessary operator actions may decrease as a result of the power uprate and the necessary flow rates for systems required to achieve and maintain safe shutdown may need to be increased. The licensee should identify the impact of the power uprate on the plant's post-fire safe-shutdown procedures.

RIS-001, Revision 0, "Review Standard for Extended Power Uprates," provides additional guidance for the review of applications for power uprate.

**SRP Section 9.5.1**
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in (Draft) Revision 4, dated October 2003 of this SRP. See ADAMS accession number ML052070563.

In addition, this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of the updated SRP section to prospective applications submitted pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 5, dated 200X:

1. Removed Branch Technical Position SPLB 9.5-1 (the BTP guidance has been incorporated in draft Revision 1 of Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," (DG-1170))

2. Expanded the review guidance for new reactors

3. Added references to applicable regulatory documents issued subsequent to Revision 4

4. Deleted Appendix A: "Supplemental Fire Protection Review Criteria for Shutdown and Decommissioned Reactors" (this guidance is provided in RG 1.191)

5. Updated guidance on the use of fire modeling and probabilistic methodologies for non-NFPA 805 plants (a new separate SRP section is being prepared for NFPA 805 plants)