Revised Brunswick NFPA 805 Transition Report,
Transition to 10 CFR 50.48(c) – NFPA 805
Performance-Based Standard for Fire Protection
for Light Water Reactor Electric Generating Plants,
2001 Edition, Transition Report,
September 19, 2013
Main Report Without Attachments

Carolina Power & Light
Brunswick Steam Electric Plant
Units 1 and 2

Transition to 10 CFR 50.48(c) - NFPA 805
Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition

Transition Report

September 19, 2013

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Executive Summary

CP&L will transition the Brunswick Steam Electric Plant (BSEP), Units 1 and 2 fire protection program to a new Risk-Informed, Performance-Based (RI-PB) alternative per 10 CFR 50.48(c) which incorporates by reference NFPA 805. The licensing basis per License Condition 2.B.(6) will be superseded.

The transition process consisted of a review and update of BSEP documentation, including the development of a Fire Probabilistic Risk Assessment (PRA) using NUREG/CR-6850 as guidance. This Transition Report summarizes the transition process and results. This Transition Report contains information:

- Required by 10 CFR 50.48(c).
- Recommended by guidance document Nuclear Energy Institute (NEI) 04-02 Revision 2 and appropriate Frequently Asked Questions (FAQs).
- Recommended by guidance document Regulatory Guide 1.205, Revision 1.

Section 4 of the Transition Report provides a summary of compliance with the following NFPA 805 requirements:

- Fundamental Fire Protection Program Elements and Minimum Design Requirements
- Nuclear Safety Performance Criteria, including:
    - Non-Power Operational Modes
    - Fire Risk Evaluations
    - Radioactive Release Performance Criteria
- Monitoring Program
- Program Documentation, Configuration Control, and Quality Assurance

Section 5 of the Transition Report provides regulatory evaluations and associated attachments, including:

- Changes to License Condition
- Changes to Technical Specifications, Orders, and Exemptions
- Determination of No Significant Hazards and evaluation of Environmental Considerations

The attachments to the Transition Report provide detail to support the transition process and results.

Attachment H contains the approved FAQs not yet incorporated into the endorsed revision of NEI 04-02. These FAQs have been used to clarify the guidance in RG 1.205, NEI 04-02, and the requirements of NFPA 805 and in the preparation of this License Amendment Request.

Acronym List

ABH	Auxiliary Boiler House
AC	Alternating Current
AC/DC	Alternating Current/Direct Current
ACLE	Allowable Combustible Load Equivalent
ADAMS	Agency wide Documents Access and Management System
ADANX	Admin – Annex Building (Security Office Building)
ADS	Automatic Depressurization System
AFEB	Alternate Fire Equipment Building
AFFF	Aqueous Film Forming Foam
AHJ	Authority having jurisdiction
ANS	American Nuclear Society
AO	Auxiliary Operator
AOG	Augmented Off-Gas Building
AOV	Air Operated Valve
APCSB	Auxiliary and Power Conversion Systems Branch
ASME	American Society of Mechanical Engineers
ASSD	Alternate Safe Shutdown
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
BGA	Brunswick Global Analysis
BKR	Breaker
BNP	Brunswick Nuclear Plant (i.e., BSEP)
BOP	Balance of Plant
BSEP	Brunswick Steam Electric Plant, Units 1 and 2
BTP	Branch Technical Position
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owner's Group
CAC	Containment Atmosphere Control

CAFTA	Computer Aided Fault Tree Analysis
CAP	Corrective Action Program
CAS	Central Alarm Station
CASBCH	Caswell Beach
CAT	Capability Category
CB	Control Building
CBDTM/THERP	Cause Based Decision Tree Method/Techniques for Human Error Rate Prediction
CBT	Computer Based Training
CBDTM	Cause Based Decision Tree Method
CC	Capability Category
CC I	Capability Category I
CCDF	Conditional Core Damage Frequency
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CDM	Current Design Method
CET	Core Exit Thermocouples
CFAST	Consolidated Model of Fire and Smoke Transport
CFD	Condensate Filter Demineralizer
CFR	Code of Federal Regulation
CGB	Cable Gripping Bushing
CLB	Chlorination Building
CLB	Current Licensing Basis
CLERP	Conditional Large Early Release Probability
CLK	Nelson Firestop CLT™ Silicone Sealant
CM	Clean Maintenance
CP&L	Carolina Power and Light
CPT	Control Power Transformers
CR3	Crystal River Unit 3 Nuclear Power Plant
CRD	Control Rod Drive
CRS	Control Room Supervisor
CS	Core Spray

CSD	Cold Shutdown
CSS	Core Spray System
CST	Condensate Storage Tank
CSW	Conventional Service Water
CTPH1	Condensate Transfer Pump House Unit 1
CW	Circwater Yard
CW	Circulating Water System
CWOD	Circulating Water Ocean Discharge
DBA	Design Basis Accidents
DBD	Design Basis Document
DC	Direct Current
DFO	Diesel Fuel Oil
DG	Diesel Generator Building
DGB	Diesel Generator Building
DID	Defense-in-Depth
DSO	Director of Site Operations
DWT	Demineralized Water Tank
EC	Engineering Change
ECCS	Emergency Core Cooling System
EDB	Equipment Database
EDG	Emergency Diesel Generator
EEE	Engineering Equivalency Evaluations
EEEE	Existing Engineering Equivalency Evaluations
EHC	Electro-Hydraulic Control
EOOS	Equipment Out-of-Service
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ERFBS	Electrical Raceway Fire Barrier Systems
ESFAS	Engineered Safeguards Actuation Signal
EY	East Yard
FC	Fire Compartment
F&O	Facts and Observations

FAQ	Frequently Asked Question
FB	Fire Brigade
FDS	Fire Dynamics Simulator
FDT	Fire Dynamics Tools
FHA	Fire Hazards Analysis
FICF	Fire Induced Circuit Failure
FIN	Fix It Now
FMEA	Failure Modes and Effects Analysis
FP	Fire Protection
FPIP	Fire Protection Initiatives Project
FPPM	Fire Protection Program Manual
FPRA	Fire Probabilistic Risk Analysis or Assessment
FRE	Fire Risk Evaluation
FSA	Fire Safety Analysis
FSAR	Final Safety Analysis Report
FSSPMD	Fire Safe Shutdown Program Manager Database
FTL	Fault Tree Logic
GDC	General Design Criterion
GL	Generic License
GPAB	Global Plant Analysis Boundary
GPM	Gallons per Minute
HCTL	Heat Capacity Temperature Limit
HEAF	High Energy Arcing Fault
HEP	Human Error Probabilities
HEPA	High Efficiency Particulate Air
HFE	Human Failure Event
HGL	Hot Gas Layer
HLP	High/Low Pressure Interface
HNP	Shearon Harris Nuclear Power Plant
HP	Health Physics
HPCI	High Pressure Coolant Injection
HPI	High Pressure Injection

HR	Human Reliability Analysis technical element from the PRA standard
HRA	Human Reliability Analysis
HRE	Higher Risk Evolutions
HRR	Heat Release Rate
HSD	Hot Shutdown
HSM	Horizontal Storage Module
HSS	High Safety Significance
HVAC	Heating, Ventilation and Air Conditioning
HX	Heat Exchanger
I&C	Instrumentation and Controls
IE	Initiating Event technical element from PRA standard
IFSN	Internal Flood Scenario Development technical element from the PRA standard
IFSO	Internal Flood Source Identification technical element from the PRA standard
IPE	Individual Plant Examination
ISB	ISFSI Storage Building
ISFSI	Independent Spent Fuel Storage Installation
KPI	Key Performance Indicator
KSF	Key Safety Function
kV	Kilovolt
kW	Kilowatt
LA	Licensing Action
LAR	License Amendment Request
LCO	Limiting Condition of Operation
LDSHD	Load Shed PRA model basic event
LERF	Large Early Release Frequency
LFS	Limiting Fire Scenario
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-site Power
LOP	Loss of Power
LOSP	Loss of Off-site Power

LPCI	Low Pressure Coolant Injection
LPI	Low Pressure Injection
LSS	Low Safety Significance
MAAP	Modular Accident Analysis Program
MAF	Manual Action Feasibility
MBOCA	Miscellaneous Buildings - Owner Controlled Area
MBPA	Miscellaneous Buildings Pre-fire Plans - Protected Area
MCA	Multiple Compartment Evaluation Approach
MCC	Motor Control Center
MCR	Main Control Room
MEFS	Maximum Expected Fire Scenario
MHIF	Multiple High Impedance Fault
MG	Motor Generator
MO	Motor Operated
MOS	Maintenance Occupancy and Storage
MOV	Motor Operated Valve
MQH	Method of McCaffrey, Quintiere, and Harkleroad
MSF	Members of the Security Force
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSO	Multiple Spurious Operation
MSR	Moisture Separator Reheater
MUD	Make-Up Demineralizer
MWT	Makeup Water Treatment Building
NCR	Nuclear Condition Report
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NEIL	Nuclear Electric Insurance Limited
NFPA	National Fire Protection Association
NFPA 805	National Fire Protection Association Standard 805
NPP	Nuclear Power Plant
NPO	Non-Power Operations

NPOPMD	Non-Power Operations Program Manager Database
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSCA	Nuclear Safety Capability Assessment
NSEL	Nuclear Safety Equipment List
NSP	Non-Suppression Probability
NSW	Nuclear Service Water
NUREG	US Nuclear Regulatory Commission Regulation
NWY	Northwest Yard
O&M	Operations and Maintenance
OCA	Owner Controlled Area
OMA	Operator Manual Action
OMB	Operations/Maintenance Building
OOS	Out-of-Service
ORAM	Outage Risk Assessment and Management
OSI PI	Real Time plant data tracking software
OSP	On-Site Power
PAM	Post-Accident Monitoring
PB	Performance Based
PBAA	Power Block Auxiliary Areas
PDC	Power Distribution Center
PFP	Pre-Fire Plans
PGM	Plant General Manager
P&ID	Piping and Instrumentation Diagram
PLC	Professional Loss Control
PMP	Pump
PNL	Panel
PNSC	Plant Nuclear Safety Committee
POM	Plant Operating Manual
PORV	Power Operated Relief Valves
POS	Plant Operational State

PRA	Probabilistic Risk Assessment or Analysis
PVC	Polyvinyl-chloride
PWR	Pressurized Water Reactor
QLS	Fire Qualitative Screening technical element from the PRA standard
QU	Quantification technical element from the PRA standard
RA	Recovery Actions
RAI	Request for Additional Information
RAW	Risk Achievement Worth
RB	Reactor Building
RBCCW	Reactor Building Closed Cooling Water
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RCR	Reactor Coolant Recirculation
RCS	Reactor Coolant System
RFP	Reactor Feed Pump
RFPT	Reactor Feed Pump Turbine
RG	Regulatory Guide
RHR	Residual Heat Removal
RI-PB	Risk-Informed Performance-Based
RIS	Regulatory Issues Summary
RMA	Radioactive Materials Area
RMCSB	Radioactive Material – Container Storage Building
RPDC	Recirc Power Distribution Center
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSDP	Remote Shutdown Panel
RW	Radwaste
RWB	Radwaste Building
RWCU	Reactor Water Cleanup System
SAMA	Severe Accident Mitigation Alternative
SAP	Secondary Access Point

SAR	Safety Analysis Report
SAT	Startup Auxiliary Transformer
SBGT	Standby Gas Treatment
SBO	Station Blackout
SCAFF	Clean Scaffold Material Storage
SCBA	Self Contained Breathing Apparatus
SD	System Description
SDC	Shutdown Cooling
SDV	Scram Discharge Volume
SE	Safety Evaluation
SER	Safety Evaluation Report
SFPC	Spent Fuel Pool Cooling
SFPE	Society of Fire Protection Engineers
SHF	Sodium Hypochlorite Facility
SIC	Site Incident Commander
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SM	Safety Margin
SP	Suppression Pool
SPC	Suppression Pool Cooling
SR	Supporting Requirement
SRV	Safety Relief Valve
SSA	Safe Shutdown Analysis
SSC	Structures, Systems, and Components
SSD	Safe Shutdown
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
STORES	Hot Shop/Material Issue/Warehouse
STORM	Storm Drain Monitoring
SW	Service Water
SWB	Service Water Building
SWGR	Switchgear

SWY	Switchyard
SY	Switchyard
TAP	Training Administrative Procedure
TB	Turbine Building
TS	Technical Specification
UAT	Unit Auxiliary Transformer
UFSAR	Updated Final Safety Analysis Report
VFDR	Variances from the deterministic requirements
VFDs	Variable Frequency Drives
V&V	Verification and Validation
WFSS	Water-based Fire Suppression System
WW	Wet well
ZOI	Zone of Influence

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) has promulgated an alternative rule for fire protection requirements at nuclear power plants, 10 CFR 50.48(c), National Fire Protection Association Standard 805 (NFPA 805). CP&L is implementing the Nuclear Energy Institute methodology NEI 04-02, "Guidance for Implementing a Risk-informed, Performance-based Fire Protection Program Under 10 CFR 50.48(c)", to transition BSEP from its current fire protection licensing basis to the new requirements as outlined in NFPA 805. This report describes the transition methodology utilized and documents how BSEP complies with the new requirements.

1.1 Background

1.1.1 NFPA 805 – Requirements and Guidance

On July 16, 2004 the NRC amended 10 CFR 50.48, Fire Protection, to add a new subsection, 10 CFR 50.48(c), which establishes new Risk-Informed, Performance-Based (RI-PB) fire protection requirements. 10 CFR 50.48(c) incorporates by reference, with exceptions, the National Fire Protection Association's NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition, as a voluntary alternative to 10 CFR 50.48 Section (b), Appendix R, and Section (f), Decommissioning.

As stated in 10 CFR 50.48(c)(3)(i), any licensee's adoption of a RI-PB program that complies with the rule is voluntary. This rule may be adopted as an acceptable alternative method for complying with either 10 CFR 50.48(b), for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979, or 10 CFR 50.48(f), plants shutdown in accordance with 10 CFR 50.82(a)(1).

NEI developed NEI 04-02 to assist licensees in adopting NFPA 805 and making the transition from their current fire protection licensing basis to one based on NFPA 805. The NRC issued Regulatory Guide (RG) 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light Water Nuclear Power Plants, which endorses NEI 04-02, with exceptions, in December 2009.[1]

A depiction of the primary document relationships is shown in Figure 1-1:

[1] Where referred to in this document NEI 04-02 is Revision 2 and RG 1.205 is Revision 1.

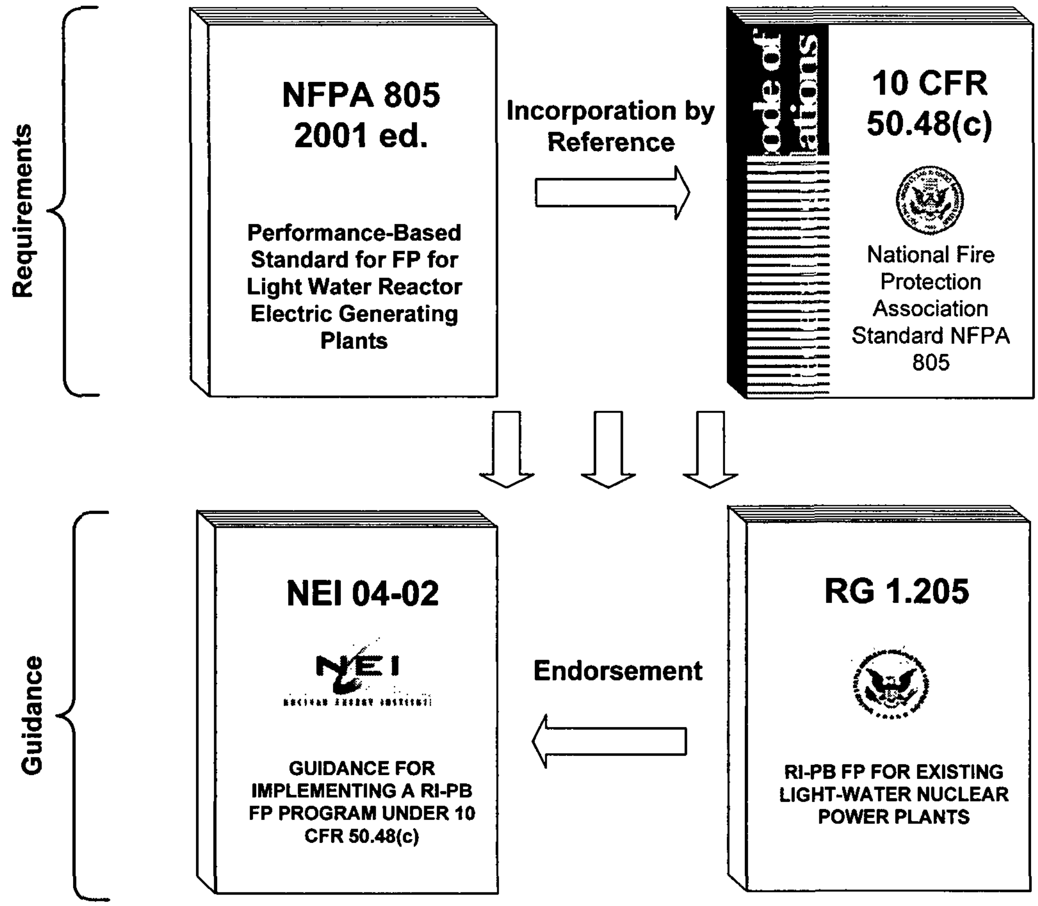

Figure 1-1 NFPA 805 Transition – Implementation Requirements/Guidance

1.1.2 Transition to 10 CFR 50.48(c)

1.1.2.1 Start of Transition

CP&L submitted a letter of intent to the NRC on June 10, 2005 (ML051720404), for the Shearon Harris Nuclear Power Plant (HNP) to adopt NFPA 805 in accordance with 10 CFR 50.48(c). This letter of intent also addressed other CP&L plants (Brunswick Steam Electric Plant Units No. 1 and 2, H.B. Robinson Steam Electric Plant Unit No. 2, and Crystal River Unit 3 Nuclear Generating Plant). The letter of intent requested three years of enforcement discretion and proposed that HNP be considered a Pilot Plant for the NFPA 805 transition process.

By letter dated April 29, 2007 (ML070590625), the NRC granted a three year enforcement discretion period. In accordance with NRC Enforcement Policy, the enforcement discretion period will continue until the NRC approval of the license amendment request (LAR) is completed.

1.1.2.2 Transition Process

The transition to NFPA 805 includes the following high level activities:

- Complete Safe Shutdown Analysis Reconstitution (activities started in 2003)

- A new Fire Probabilistic Risk Assessment (PRA) using NUREG/CR-6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, as guidance and a revision to the Internal Events PRAs to support the Fire PRAs
- Completion of activities required to transition the pre-transition licensing basis to 10 CFR 50.48(c) as specified in NEI 04-02 and RG 1.205

The project was implemented using a comprehensive project plan and individual procedures/instructions for individual scopes of work. These procedures/instructions (e.g., Project Instruction "FPIP" series procedures referenced in this report) were developed for the purposes of NFPA 805 transition. Appropriate technical content from these procedures were and will be incorporated into technical documents and configuration control procedures, as required.

1.2 Purpose

The purpose of the Transition Report is as follows:

1) Describe the process implemented to transition the current fire protection program to comply with the additional requirements of 10 CFR 50.48(c).
2) Summarize the results of the transition process.
3) Explain the bases for conclusions that the fire protection program complies with 10 CFR 50.48(c) requirements.
4) Describe the new fire protection licensing basis.
5) Describe the configuration management processes used to manage post-transition changes to the station and the fire protection program, and resulting impact on the licensing basis.

2.0 OVERVIEW OF EXISTING FIRE PROTECTION PROGRAM

2.1 Current Fire Protection Licensing Basis

Brunswick Steam Electric Plant was licensed to operate on September 8, 1976, for Unit 1 and December 27, 1974, for Unit 2. As a result, the Brunswick Steam Electric Plant fire protection program is based on evaluation and NRC acceptance against the requirements of Design Criterion 3, Appendix A to 10 CFR 50 Part 50, and 10 CFR 50 Appendix R, Sections III.G and J. The following License Condition 2.B(6) in Amendment No. 169 to the Facility Operating License No. DPR-71 (i.e., Unit 1) and Amendment No. 200 to Facility Operating License No. DPR-62 (i.e., Unit 2) states:

> *"Carolina Power and Light Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report, dated November 22, 1977, as supplemented April 1979, June 11, 1980, December 30, 1986, December 6, 1989, and July 28, 1993 and February 10, 1994, respectively, subject to the following provision:*
>
> > *The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire."*

2.2 NRC Acceptance of the Fire Protection Licensing Basis

The Commission issued, on November 22, 1977, Amendment No. 11 to the Facility Operating License No. DPR-71, for Unit No. 1, and Amendment No. 37 to Facility Operating License No. DPR-62, for Unit No. 2, of the Brunswick Steam Electric Plant. These amendments added license conditions relating to the completion of the facility modifications for fire protection and resolution of incomplete items. The amendment for Unit 1 also incorporated limiting conditions for operations and surveillance requirements for existing fire protection systems and administrative controls.

Amendment No. 11 contained the following changes to 2.B(6) and 2.C.(2):

2.B(6) The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1 through 3.1.35 of the NRC's Fire Protection Safety Evaluation Report on the Brunswick facility dated November 22, 1977. These modifications shall be completed by the end of the first refueling outage of Brunswick Unit 1 and prior to return to operation of Cycle 2. In addition, the licensee shall submit the additional information identified in Table 3.1 of this Safety Evaluation Report in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report explaining the circumstances, together with a revised schedule.

2.C.(2) The Technical Specifications contained in Appendices A, A-Prime and B, attached hereto, as revised through Amendment No. 11, are hereby incorporated in this license. Appendix A shall be effective from the date of issuance of the Unit 1 operating license until the Appendix A-Prime becomes effective on or before the initial criticality of Brunswick Unit 2 following its initial refueling outage. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications as indicated above. The licensee shall inform the Office of Inspection and Enforcement, Region II, of the date that the Appendix A-Prime becomes effective.

Amendment No. 37 contained the following changes to 2.B(7):

2.B(7) The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1 through 3.1.35 of the NRC's Fire Protection Safety Evaluation Report on the Brunswick facility dated November 22, 1977. These modifications shall be completed by the end of the second refueling outage of Brunswick Unit 2 and prior to return to operation of Cycle 3. In addition, the licensee shall submit the additional information identified in Table 3.1 of this Safety Evaluation Report in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report explaining the circumstances, together with a revised schedule.

The Commission issued, on April 6, 1979, Amendment No. 23 to the Facility Operating License No. DPR-71, for Unit No. 1, and Amendment No. 47 to Facility Operating License No. DPR-62, for Unit No. 2, of the Brunswick Steam Electric Plant. These amendments consisted of changes to the operating licenses for both units to allow revised implementation dates for certain modifications intended to improve the level of fire protection. Supplement 1 of the Fire Protection Safety Evaluation Report was also included in this transmittal which addressed certain items that were identified as incomplete and requiring further information from the licensee and evaluation by the staff. The SER, Supplement 1, also listed several modifications proposed by the licensee to improve fire protection.

The Commission issued Supplement 2 to the Fire Protection Safety Evaluation Report on June 11,1980, which contained evaluations associated with four areas: 1) Protection of Redundant Safe Shutdown Cabling (greater than five foot separation, 2) Protection of Redundant Safe Shutdown Cabling (less than five foot separation), 3) Fire Protection Loop Isolation Valve, 4) Door Frames for Fire Doors.

The Commission granted, on November 10, 1981, an exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.3, with regard to fixed fire suppression in the Control Room.

The Commission granted, on July 27, 1983, exemptions from the requirements of 10 CFR 50 Appendix R, Section III.G.3, with regard to fixed fire suppression in the seven fire zones in the Control Building Cable Vaults.

The Commission granted, on September 17, 1986, an exemption from the licensee commitment to install an excess flow switch and automatic shut-off valve in the fuel supply line for the diesel fire pump to automatically isolate in the event of a fuel line rupture. The Safety Evaluation concluded that the previous commitment to provide automatic isolation of the diesel fuel line need not be implemented because of the alternative fire protection measures provided.

The Commission granted, on December 30, 1986, exemptions from the requirements of Appendix R to 10 CFR Part 50, Sections III.G and J. Exemptions were granted for the following:

1) Reactor Buildings, Units 1 and 2 (Fire Areas RB1-1 and RB2-1)
2) Emergency Core Cooling System Rooms, Units 1 and 2 (Fire Areas RB1-6 and RB2-6)
3) Diesel Generator Building Basement (Fire Area DG-1)
4) Service Water Building (Fire Area SW-1)
5) Diesel Generator Building (DG-08)
6) Fixed Fire Suppression System for Alternative Shutdown Areas (Fire Areas TB-1, CB-1, CB-7, CB-8, CB-9, CB-10, DG-6, DG-7, DG-9, DG-11, DG-12, DG-13 and DG-14)
7) East Yard Area

The Staff concluded that the exemption request for the Control Building Extended (Fire Area CB-23E) was not needed.

The Commission issued, on May 29, 1987, a Safety Evaluation approving the use of higher unexposed side temperatures for fire barrier seals than that required by the Branch Technical Position (BTP) ASB 9.5-1 of NUREG-0800. The evaluation concluded that the acceptance criteria of 325 °F above ambient, versus 250 °F above ambient, was an acceptable deviation and was not considered likely to significantly add to the risk of igniting material on the unexposed side of the barrier.

The Commission granted, on August 27, 1987, an Exemption from 10 CFR Part 50, Appendix R, Section III.J, from the requirement for emergency lighting units with at least an 8-hour battery supply in all areas needed for operation of safe shutdown equipment. The exemption permits substitution of 8-hour battery lighting with:

1) The use of diesel generators to power lighting in the plant control room upon loss of offsite power.
2) The use of two-hour battery-powered lighting upon loss of diesel generators concurrent with loss of offsite power.
3) Assurance that power sources are routed underground and are separated by at least a three-hour rated fire barrier.

The Commission issued an Appendix R Safety Evaluation Clarification and Revision on December 6, 1989. Brunswick Steam Electric Plant had identified nineteen items associated with the Staff's December 30, 1986, Safety Evaluation where revisions were required to 1) correct specific errors, 2) clarify potentially confusing language, or 3) more accurately state actual conditions. The Staff provided clarifications for fifteen of the nineteen items requested. These clarifications appended the December 30, 1986, Safety Evaluation.

The Commission issued, on July 28, 1993, a Safety Evaluation approving a request to downgrade the three-hour rated masonry block walls in the control building cable access ways (separating fire areas CB-01a/b, CB-02a/b, CB-12a/b and CB-13a/b) to non-rated walls. The Staff found this change did not have an adverse impact on the III.G.3 exemption granted for the lack of fire suppression in the control building and would not impact the alternate shutdown capability.

The Commission issued, on February 10, 1994, a Safety Evaluation that revised the plant fire protection licensing condition and Technical Specifications (TS). In accordance with Generic Letter 86-10 and 88-12, CP&L requested that fire protection be removed from the Technical Specifications and a standard fire protection licensing condition be implemented. The following Technical Specification changes were proposed and granted by the NRC:

1) Delete TS 3.3.5.7 (Fire Detection Instrumentation), TS 3.7.7.1 (Fire Suppression Water System), TS 3.7.7.2 (Spray and/or Sprinkler Systems), TS 3.7.7.3, (High Pressure Carbon Dioxide), TS 3.7.7.4 (Fire Hose Stations), TS 3.7.7.5, (Foam Systems), and TS 3.7.8 (Fire Barrier Penetrations) and their associated bases and incorporate into the Updated Final Safety Analysis Report (UFSAR).
2) Delete TS 6.2.2.g for site fire brigade staffing and incorporate into the UFSAR
3) Delete TS 6.4.2 requirements related to the fire brigade training program and incorporate into the UFSAR.
4) Add TS 6.5.3.8(m) to include the review of the fire protection program and implementing procedures as an additional responsibility of the Plant Nuclear Safety Committee (PNSC).
5) Delete TS 6.9.2.d related to the requirement for special reports for the fire detection instrumentation.
6) Delete TS 6.9.2.g related to the requirement for special reports for fire suppression systems.
7) Delete TS 6.9.2.h related to the requirement for special reports for fire barrier penetrations.

CP&L also proposed, and the NRC granted the request, to replace the current fire protection licensee condition with the standard license condition provided in GL 86-10.

3.0 TRANSITION PROCESS

3.1 Background

Section 4.0 of NEI 04-02 describes the process for transitioning from compliance with the current fire protection licensing basis to the new requirements of 10 CFR 50.48(c). NEI 04-02 contains the following steps:

1) Licensee determination to transition the licensing basis and devote the necessary resources to it;
2) Submit a Letter of Intent to the NRC stating the licensee's intention to transition the licensing basis in accordance with a tentative schedule;
3) Conduct the transition process to determine the extent to which the current fire protection licensing basis supports compliance with the new requirements and the extent to which additional analyses, plant and program changes, and alternative methods and analytical approaches are needed;
4) Submit a LAR;
5) Complete transition activities that can be completed prior to the receipt of the License Amendment;
6) Receive a Safety Evaluation; and
7) Complete implementation of the new licensing basis, including completion of modifications identified in Attachment S.

3.2 NFPA 805 Process

Section 2.2 of NFPA 805 establishes the general process for demonstrating compliance with NFPA 805. This process is illustrated in Figure 3-1. It shows that, except for the fundamental fire protection requirements, compliance can be achieved on a fire area basis either by deterministic or RI-PB methods. Consistent with the guidance in NEI 04-02, CP&L has implemented the NFPA 805, Section 2.2 process, by first determining the extent to which its current fire protection program supports findings of deterministic compliance with the requirements in NFPA 805. RI-PB methods are being applied to the requirements for which deterministic compliance could not be shown.

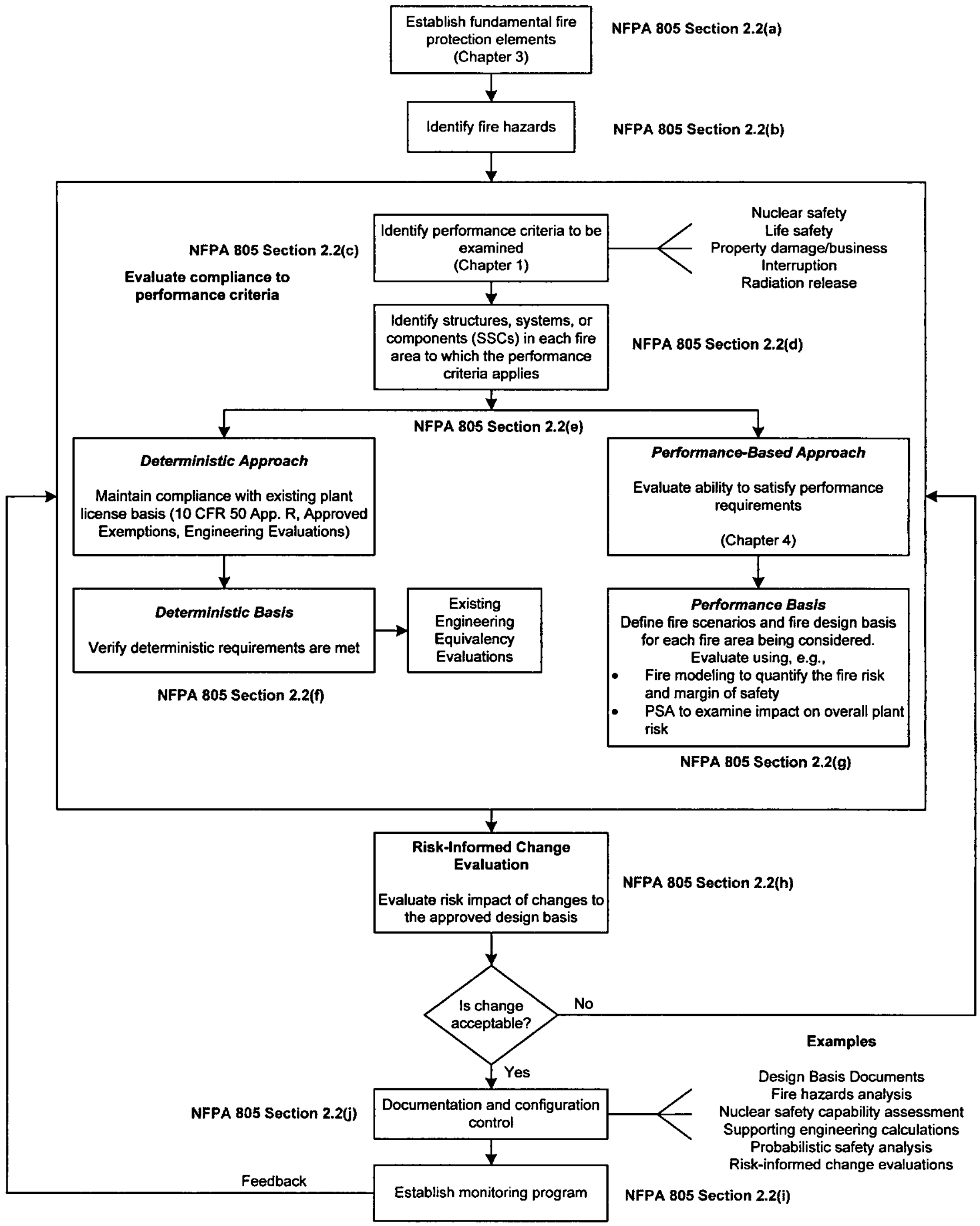

Figure 3-1 NFPA 805 Process [NEI 04-02 Figure 3-1 based on Figure 2-2 of NFPA 805][2]

[2] Note: 10 CFR 50.48(c) does not incorporate by reference Life Safety and Plant Damage/Business Interruption goals, objectives and criteria. See 10 CFR 50.48(c) for specific exceptions to the incorporation by reference of NFPA 805.

3.3 NEI 04-02 – NFPA 805 Transition Process

NFPA 805 contains technical processes and requirements for a RI-PB fire protection program. NEI 04-02 was developed to provide guidance on the overall process (i.e., programmatic, technical, and licensing) for transitioning from a traditional fire protection licensing basis to a new RI-PB method based upon NFPA 805, as shown in Figure 3-2.

Section 4.0 of NEI 04-02 describes the detailed process for assessing a fire protection program for compliance with NFPA 805, as shown in Figure 3-2.

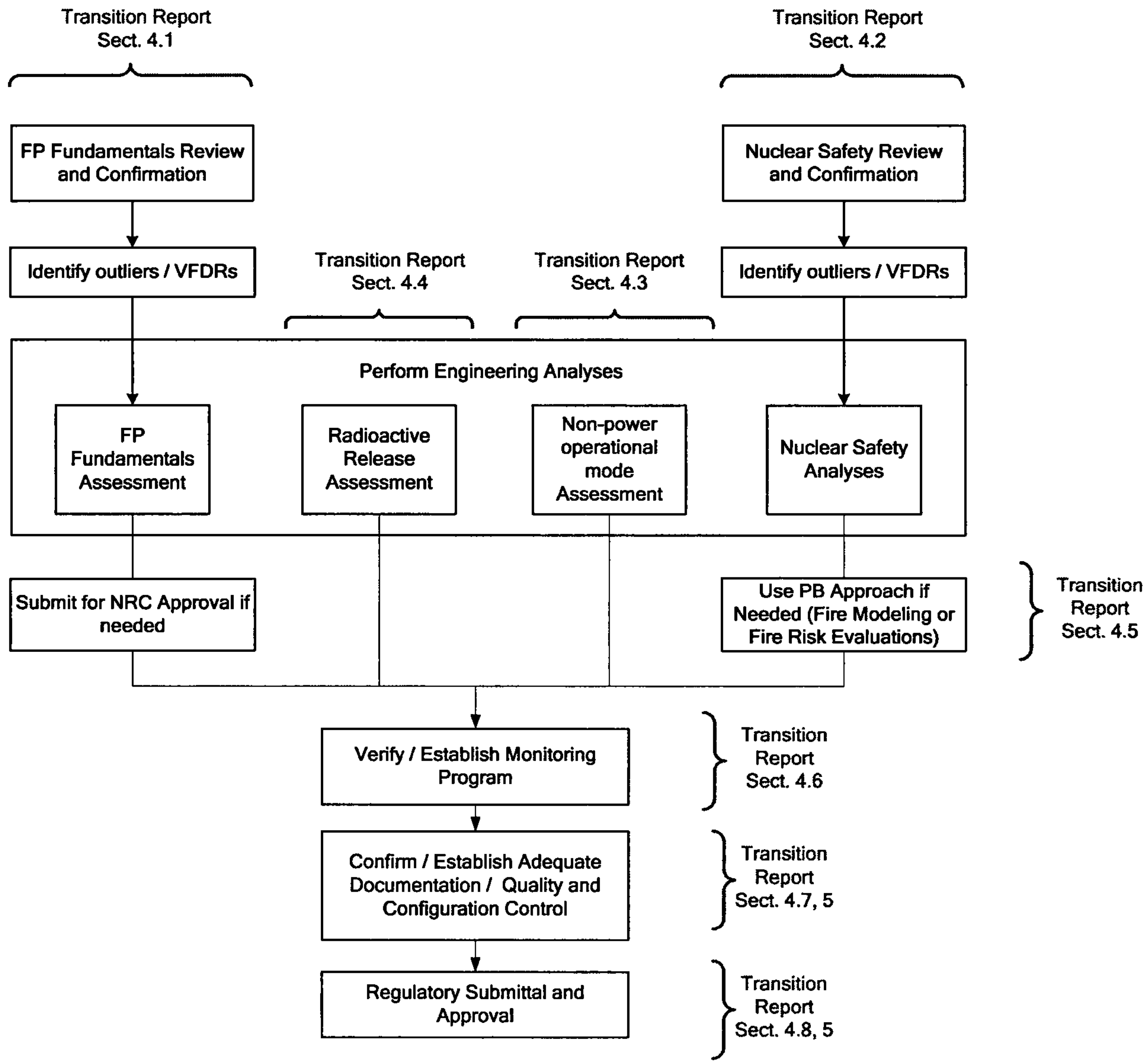

Figure 3-2 Transition Process (Simplified) [based on NEI 04-02 Figure 4-1]

3.4 NFPA 805 Frequently Asked Questions (FAQs)

The NRC has worked with NEI and two Pilot Plants (Oconee Nuclear Station and HNP) to define the licensing process for transitioning to a new licensing basis under 10 CFR 50.48(c) and NFPA 805. Both the NRC and the industry recognized the need for additional clarifications to the guidance provided in RG 1.205, NEI 04-02, and the requirements of NFPA 805. The NFPA 805 FAQ process was jointly developed by NEI and NRC to facilitate timely clarifications of NRC positions. This process is described in a letter from the NRC dated July 12, 2006, to NEI (ML061660105) and in Regulatory Issue Summary (RIS) 2007-19, Process for Communicating Clarifications of Staff Positions Provided in RG 1.205 Concerning Issues Identified during the Pilot Application of NFPA Standard 805, dated August 20, 2007 (ML071590227).

Under the FAQ Process, transition issues are submitted to the NEI NFPA 805 Task Force for review, and subsequently presented to the NRC during public FAQ meetings. Once the NEI NFPA 805 Task Force and NRC reach agreement, the NRC issues a memorandum to indicate that the FAQ is acceptable. NEI 04-02 will be revised to incorporate the approved FAQs. This is an on-going revision process that will continue through the transition of NFPA 805 plants. Final closure of the FAQs will occur when future revisions of RG 1.205, endorsing the related revisions of NEI 04-02, are approved by the NRC. It is expected that additional FAQs will be written, and existing FAQs will be revised, as plants continue NFPA 805 transition after the Pilot Plant Safety Evaluations.

Attachment H contains the list of approved FAQs not yet incorporated into the endorsed revision of NEI 04-02. These FAQs have been used to clarify the guidance in RG 1.205, NEI 04-02, and the requirements of NFPA 805 and in the preparation of this LAR.

4.0 COMPLIANCE WITH NFPA 805 REQUIREMENTS

4.1 Fundamental Fire Protection Program and Design Elements

The Fundamental Fire Protection Program and Design Elements are established in Chapter 3 of NFPA 805. Section 4.3.1 of NEI 04-02 provides a systematic process for determining the extent to which the pre-transition licensing basis and plant configuration meets these criteria and for identifying the fire protection program changes that would be necessary for compliance with NFPA 805. NEI 04-02 Appendix B-1 provides guidance on documenting compliance with the program requirements of NFPA 805 Chapter 3.

4.1.1 Overview of Evaluation Process

The comparison of the BSEP Fire Protection Program to the requirements of NFPA 805 Chapter 3 was performed and documented in Attachment A, Table B-1, NFPA 805 Ch. 3 Transition Details. The analysis used the guidance contained in NEI 04-02, Section 4.3.1 and Appendix B-1 (See Figure 4-1).

Each section and subsection of NFPA 805 Chapter 3 was reviewed against the current fire protection program. Upon completion of the activities associated with the review, the following compliance statement(s) was used:

- Complies - For those sections/subsections determined to meet the specific requirements of NFPA 805.
- Complies with Clarification - For those sections/subsections determined to meet the requirements of NFPA 805 with clarification.
- Complies by previous NRC approval - For those sections/subsections where the specific NFPA 805 Chapter 3 requirements are not met but previous NRC approval of the configuration exists.
- Complies with use of Existing Engineering Equivalency Evaluations (EEEEs) - For those sections/subsections determined to be equivalent to the NFPA 805 Chapter 3 requirements as documented by engineering analysis.
- Submit for NRC Approval - For those sections/subsections for which approval is sought in this LAR submittal in accordance with 10 CFR 50.48(c)(2)(vii). A summary of the bases of acceptability is provided (see Attachment L for details).

In some cases, multiple compliance statements have been assigned to a specific NFPA 805 Chapter 3 section/subsection. Where this is the case, each compliance/compliance basis statement clearly references the corresponding requirement of NFPA 805 Chapter 3.

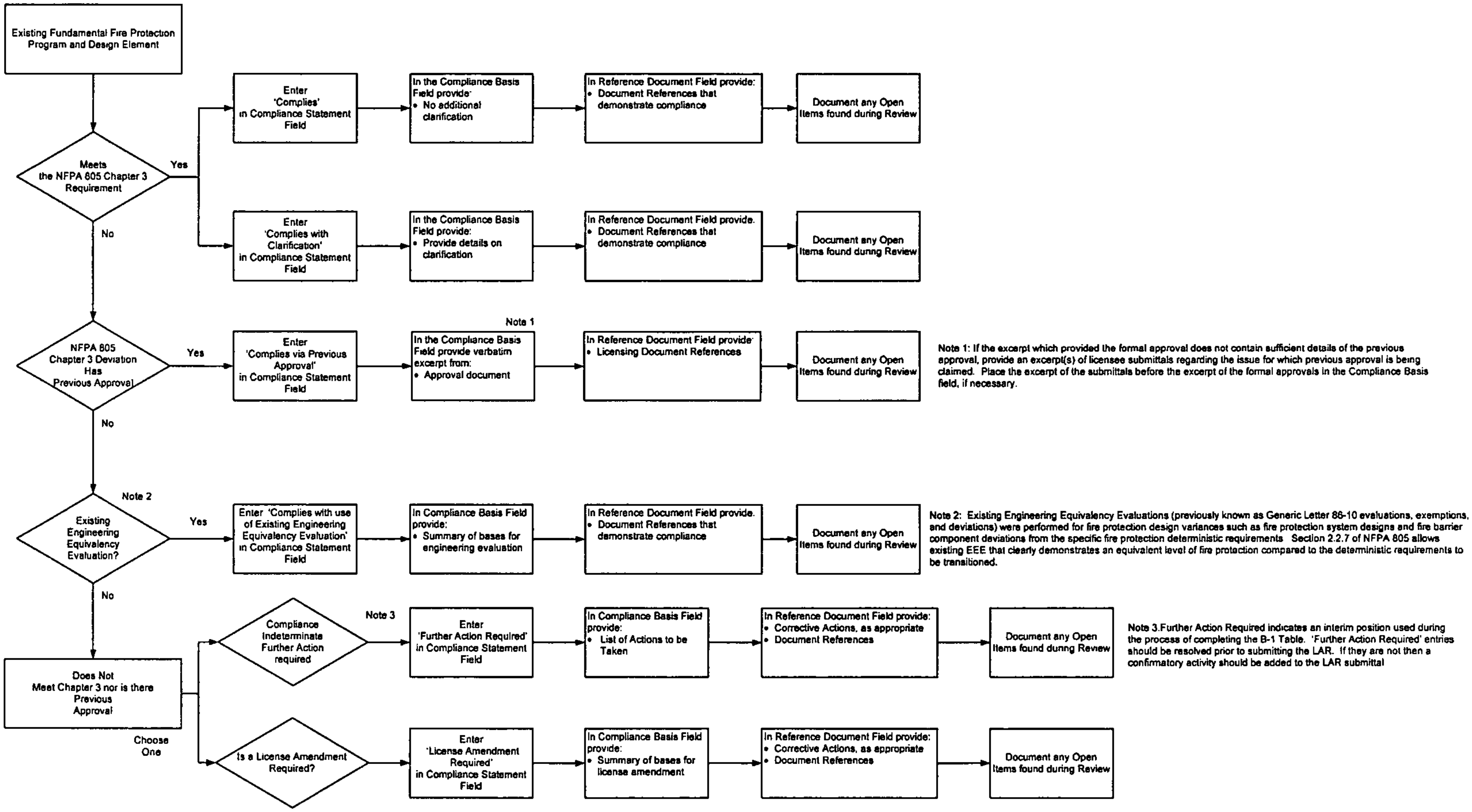

Figure 4-1 - Fundamental Fire Protection Program and Design Elements Transition Process [Based on NEI 04-02 Figure 4-2][3]

[3] Figure 4-1 depicts the process used during the transition and therefore contains elements (i.e., open items) that represent interim resolutions. Additional detail on the transition of EEEEs is included in Section 4.2.2.

4.1.2 Results of the Evaluation Process

4.1.2.1 NFPA 805 Chapter 3 Requirements Met or Previously Approved by the NRC

Attachment A contains the NEI 04-02, Table B-1, Transition of Fundamental Fire Protection Program and Design Elements. This table provides the compliance basis for the requirements in NFPA 805 Chapter 3. Except as identified in Section 4.1.2.3, Attachment A demonstrates that the fire protection program at BSEP either:

- Complies directly with the requirements of NFPA 805 Chapter 3,
- Complies with clarification with the requirements of NFPA 805 Chapter 3,
- Complies through the use of existing engineering equivalency evaluations which are valid and of appropriate quality, or
- Complies with a previously NRC approved alternative to NFPA 805 Chapter 3 and, therefore, the specific requirement of NFPA 805 Chapter 3 is supplanted.

4.1.2.2 NFPA 805 Chapter 3 Requirements Requiring Clarification of Prior NRC Approval

NPFA 805 Section 3.1 states in part, "Previously approved alternatives from the fundamental protection program attributes of this chapter by the AHJ take precedence over the requirements contained herein." In some cases, prior NRC approval of an NFPA 805 Chapter 3 program attribute may be unclear. CP&L requests that the NRC concur with their finding of prior approval for the following sections of NFPA 805 Chapter 3:

- None.

4.1.2.3 NFPA 805 Chapter 3 Requirements Not Previously Approved by NRC

The following sections of NFPA 805, Chapter 3, are not specifically met nor do previous NRC approvals of alternatives exist:

- 3.5.16 – Approval is requested for the use of fire protection water for specific plant evolutions.
- 3.2.3(1) – Approval is requested for the use of performance-based methods to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features required by NFPA 805.

The specific deviation and a discussion of how the alternative satisfies 10 CFR 50.48(c)(2)(vii) requirements are provided in Attachment L. CP&L requests NRC approval of these performance-based methods.

4.1.3 Definition of Power Block and Plant

Where used in NFPA 805, Chapter 3, the terms "Power Block" and "Plant" refer to structures that have equipment required for nuclear plant operations, such as Containment, Auxiliary Building, Service Building, Control Building, Fuel Building, Radioactive Waste, Water Treatment, Turbine Building, and intake structures or structures that are identified in the facility's pre-transition licensing basis.

All structures within the BSEP Owner Controlled Area were reviewed to determine the potential impact of fire on the nuclear safety and radioactive release criteria described in Section 1.5 of NFPA 805. This was accomplished by identifying the structures that contain either

- Equipment that could affect
    - Plant operation for power generation
    - Ability to maintain nuclear safety performance criteria in the event of a fire, including Safe Shutdown Capability

OR

- Radioactive materials that could potentially be released in event of a fire

These structures are listed in Attachment I and define the "power block" and "plant".

4.2 Nuclear Safety Performance Criteria

The Nuclear Safety Performance Criteria are established in Section 1.5 of NFPA 805. Chapter 4 of NFPA 805 provides the methodology to determine the fire protection systems and features required to achieve the performance criteria outlined in Section 1.5. Section 4.3.2 of NEI 04-02 provides a systematic process for determining the extent to which the pre-transition licensing basis meets these criteria and for identifying any necessary fire protection program changes. NEI 04-02, Appendix B-2, provides guidance on documenting the transition of Nuclear Safety Capability Assessment Methodology and the Fire Area compliance strategies.

4.2.1 Nuclear Safety Capability Assessment Methodology

The Nuclear Safety Capability Assessment (NSCA) Methodology review consists of four processes:

- Establishing compliance with NFPA 805 Section 2.4.2
- Establishing the Safe and Stable Conditions for the Plant
- Establishing Recovery Actions
- Evaluating Multiple Spurious Operations

The methodology for demonstrating reasonable assurance that a fire during non-power operational (NPO) modes will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition is an additional requirement of 10 CFR 50.48(c) and is addressed in Section 4.3.

4.2.1.1 Compliance with NFPA 805 Section 2.4.2

Overview of Process

NFPA 805 Section 2.4.2 Nuclear Safety Capability Assessment states:

> *"The purpose of this section is to define the methodology for performing a nuclear safety capability assessment. The following steps shall be performed:*
> *(1) Selection of systems and equipment and their interrelationships necessary to achieve the nuclear safety performance criteria in Chapter 1*
> *(2) Selection of cables necessary to achieve the nuclear safety performance criteria in Chapter 1*
> *(3) Identification of the location of nuclear safety equipment and cables*
> *(4) Assessment of the ability to achieve the nuclear safety performance criteria given a fire in each fire area"*

The NSCA methodology review evaluated the NSCA methodology against the guidance provided in NEI 00-01, Revision 1 (ML050310295) Chapter 3, "Deterministic Methodology," as discussed in Appendix B-2 of NEI 04-02. The methodology is depicted in Figure 4-2 and consisted of the following activities:

- Each specific section of NFPA 805 2.4.2 was correlated to the corresponding section of Chapter 3 of NEI 00-01 Revision 1. Based upon the content of the NEI 00-01 methodology statements, a determination was made of the applicability of the section to the station.
- The plant-specific methodology was compared to applicable sections of NEI 00-01 and one of the following alignment statements and its associated basis were assigned to the section:
    - Aligns
    - Aligns with intent
    - Not in Alignment
    - Not in Alignment, but Prior NRC Approval
    - Not in Alignment, but no adverse consequences
- For those sections that do not align, an assessment was made to determine if the failure to maintain strict alignment with the guidance in NEI 00-01 could have adverse consequences. Since NEI 00-01 is a guidance document, portions of its text could be interpreted as 'good practice' or intended as an example of an efficient means of performing the analyses. If the section has no adverse consequences, these sections of NEI 00-01 can be dispositioned without further review.

The comparison of the BSEP existing post-fire Safe Shutdown Analysis (SSA) methodology to NEI 00-01 Chapter 3 (NEI 04-02 Table B-2) was performed and documented in Attachment B, Table B-2 Nuclear Safety Capability Assessment Methodology Review.

In addition, a review of NEI 00-01, Revision 2, (ML091770265) Chapter 3, was conducted to identify the substantive changes from NEI 00-01, Revision 1 that are applicable to an NFPA 805 fire protection program. This review was performed and documented in Attachment B, Table B-2 Nuclear Safety Capability Assessment Methodology Review.

Results from Evaluation Process

The method used to perform the existing post-fire SSA with respect to selection of systems and equipment, selection of cables, and identification of the location of equipment and cables, either meets the NRC endorsed guidance from NEI 00-01, Revision 1, Chapter 3 (i.e., as supplemented by the gap analysis) directly or met the intent of the endorsed guidance with adequate justification as documented in Attachment B.

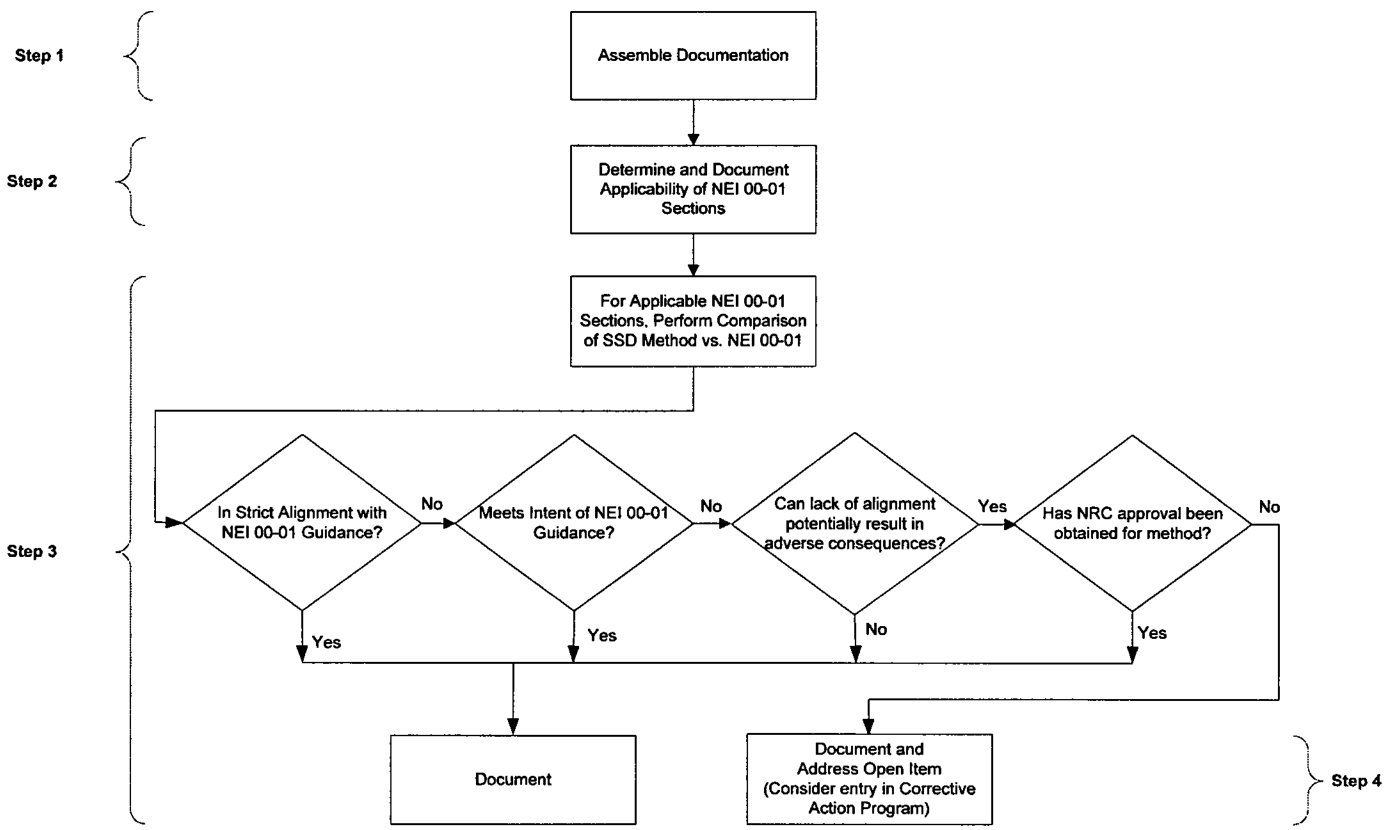

Figure 4-2 – Summary of Nuclear Safety Methodology Review Process (FAQ 07-0039)

Comparison to NEI 00-01 Revision 2

An additional review was performed of NEI 00-01, Revision 2, Chapter 3, for specific substantive changes in the guidance from NEI 00-01, Revision 1 that are applicable to an NFPA 805 transition. The results of this review are summarized below:

- Post fire manual operation of rising stem valves in the fire area of concern (NEI 00-01 Section 3.2.1.2)

  A review of the NSCA results indicated that there are no recovery actions or defense-in-depth recovery actions that require manual operation of a rising stem valve in the fire area of concern.
- Analysis of open circuits on a high voltage (e.g., 4.16 kV) ammeter current transformers (NEI 00-01 Section 3.5.2.1)

  The evaluation concludes that this failure mode is unlikely for CTs that could pose a threat to safe shutdown equipment.
- Analysis of control power for switchgear with respect to breaker coordination (NEI 00-01 Section 3.5.2.4)

  Control power is modeled in the safe shutdown fault tree used to develop the NSCA. A loss of control power results in an assumed loss of the switchgear, and there are no cases where a bus is credited to remain operable without control power.

4.2.1.2 Safe and Stable Conditions for the Plant

Overview of Process

The nuclear safety goals, objectives and performance criteria of NFPA 805 allow more flexibility than the previous deterministic programs based on 10 CFR 50 Appendix R and NUREG 0800, Section 9.5-1 (and NEI 00-01, Chapter 3) since NFPA 805 only requires the licensee to maintain the fuel in a safe and stable condition rather than achieve and maintain cold shutdown.

NFPA 805, Section 1.6.56, defines Safe and Stable Conditions as follows

> *"For fuel in the reactor vessel, head on and tensioned, safe and stable conditions are defined as the ability to maintain Keff <0.99, with a reactor coolant temperature at or below the requirements for hot shutdown for a boiling water reactor and hot standby for a pressurized water reactor. For all other configurations, safe and stable conditions are defined as maintaining Keff <0.99 and fuel coolant temperature below boiling."*

The nuclear safety goal of NFPA 805 requires "*...reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition*" without a specific reference to a mission time or event coping duration.

For the plant to be in a safe and stable condition, it may not be necessary to perform a transition to cold shutdown as currently required under 10 CFR 50, Appendix R. Therefore, the unit may remain at or below the temperature defined by a hot standby/hot shutdown plant operating state for the event.

Results

Based on the criteria discussed in NCSA calculation BNP-E-9.010, "Safe Shutdown Analysis in Case of Fire," the NFPA 805 licensing basis for BSEP is to achieve and maintain hot shutdown conditions following any fire occurring prior to establishing cold shutdown. Specifically, the conditions include:

- the reactor operating at power,
- a shutdown immediately prior to aligning the RHR system for shutdown cooling, or
- the "transition" mode between these two operational phases.

Immediately following the reactor scram, RCS inventory and pressure control is maintained using the high pressure systems, HPCI and RCIC, or the low pressure injection systems using the SRV's for pressure reduction, which are the RHR System in LPCI mode or Core Spray System. For the most limiting fire scenarios in every fire area, BNP-E-9.010 documents the availability of long term cooling using the RHR system, in either the Normal Shutdown Cooling Mode or Alternate Shutdown Cooling Mode, or the Core Spray System, all of which are characterized by low pressure injection and at least 1 SRV available to provide core flow. The RHR Service Water system rejects decay heat to the ultimate heat sink.

Notably, initiation of RHR in the suppression pool cooling mode does not imply that the plant would proceed all the way to cold shutdown. Following stabilization at hot shutdown, a long term strategy for decay heat removal and inventory/pressure control would be determined based on the extent of equipment damage. If an assessment of the post-fire conditions indicated that placing RHR in the Shutdown Cooling or Alternate Shutdown Cooling modes would be advisable, then activities would commence in a safe and controlled manner to align plant equipment required for reactor cooldown.

The long-term actions required to maintain safe and stable conditions are relatively low risk activities that are largely routine and within the normal capabilities of site personnel, even in the face of fire damage, due to the assured availability of at least one train of RHR and either onsite or offsite power sources. Repairs to safe shutdown equipment would not be required and the management of the onsite inventories of makeup water, nitrogen and diesel fuel would not require resources beyond those available from normal operations staff and emergency response personnel.

Demonstration of the Nuclear Safety Performance Criteria for safe and stable conditions was performed in two analyses.

- At-Power analysis, Modes 1-3. This analysis is discussed in Section 4.2.4.
- Non-Power Operations analysis that includes cold shutdown and below, or Modes 4 and 5. This analysis is discussed in Section 4.3.

4.2.1.3 Establishing Recovery Actions

Overview of Process

NEI 04-02 and RG 1.205 suggest that a licensee submit a summary of its approach for addressing the transition of OMAs as recovery actions in the LAR (i.e., Regulatory Position 2.2.1 and NEI-04-02, Section 4.6). As a minimum, NEI 04-02 suggests that the assumptions, criteria, methodology, and overall results be included for the NRC to determine the acceptability of the licensee's methodology.

The discussion below provides the methodology used to transition pre-transition OMAs and to determine the population of post-transition recovery actions. This process is based on FAQ 07-0030 (ML110070485) and consists of the following steps:

- Step 1: Clearly define the primary control station(s) and determine which pre-transition OMAs are taken at primary control station(s) Activities that occur in the Main Control Room are not considered pre-transition OMAs. Activities that take place at primary control station(s) or in the Main Control Room are not recovery actions, by definition.
- Step 2: Determine the population of recovery actions that are required to resolve variances from deterministic requirements (VFDRs) (i.e., to meet the risk acceptance criteria or maintain a sufficient level of defense-in-depth).
- Step 3: Evaluate the additional risk presented by the use of recovery actions required to demonstrate the availability of a success path
- Step 4: Evaluate the feasibility of the recovery actions
- Step 5: Evaluate the reliability of the recovery actions

Results

The review results are documented in the Fire Safety Analysis for each area. Refer to Attachment G for the detailed evaluation process and summary of the results from the process.

4.2.1.4 Evaluation of Multiple Spurious Operations

Overview of Process

NEI 04-02 suggests that a licensee submit a summary of its approach for addressing potential fire-induced MSOs for NRC review and approval. As a minimum, NEI 04-02 recommends that the summary contain sufficient information relevant to methods, tools, and acceptance criteria used to enable the NRC to determine the acceptability of the licensee's methodology. The methodology used to address MSOs for Brunswick is summarized below.

As part of the NFPA 805 transition project, a review and evaluation of Brunswick susceptibility to fire-induced MSOs was performed. The process was conducted in accordance with NEI 04-02 and RG 1.205, as supplemented by FAQ 07-0038 Revision 3 (ML110140242). The BWR Generic MSO list from NEI 00-01, Revision 3 was utilized.

The approach outlined in Figure 4-3, below, (i.e., based on Figure 4-8 from FAQ 07-0038) is the method used to address fire-induced MSOs for BSEP. This method used insights from the Fire PRA developed in support of transition to NFPA 805 and consists of the following:

- Identifying potential MSOs of concern.
- Conducting an expert panel to assess plant specific vulnerabilities (e.g., per NEI 00-01, Rev. 1 Section F.4.2).
- Updating the Fire PRA model and existing post-fire SSA / NSCA to include the MSOs of concern.
- Evaluating for NFPA 805 compliance.
- Documenting results.

This process is intended to support the transition to a new licensing basis. Post-transition changes would use the RI-PB change process. The post-transition change process for the assessment of a specific MSO would be a simplified version of this process, and may not need the level of detail shown in the following section (e.g., an expert panel may not be necessary to identify and assess a new potential MSO. Identification of new potential MSOs will be part of the plant change review process and/or inspection process).

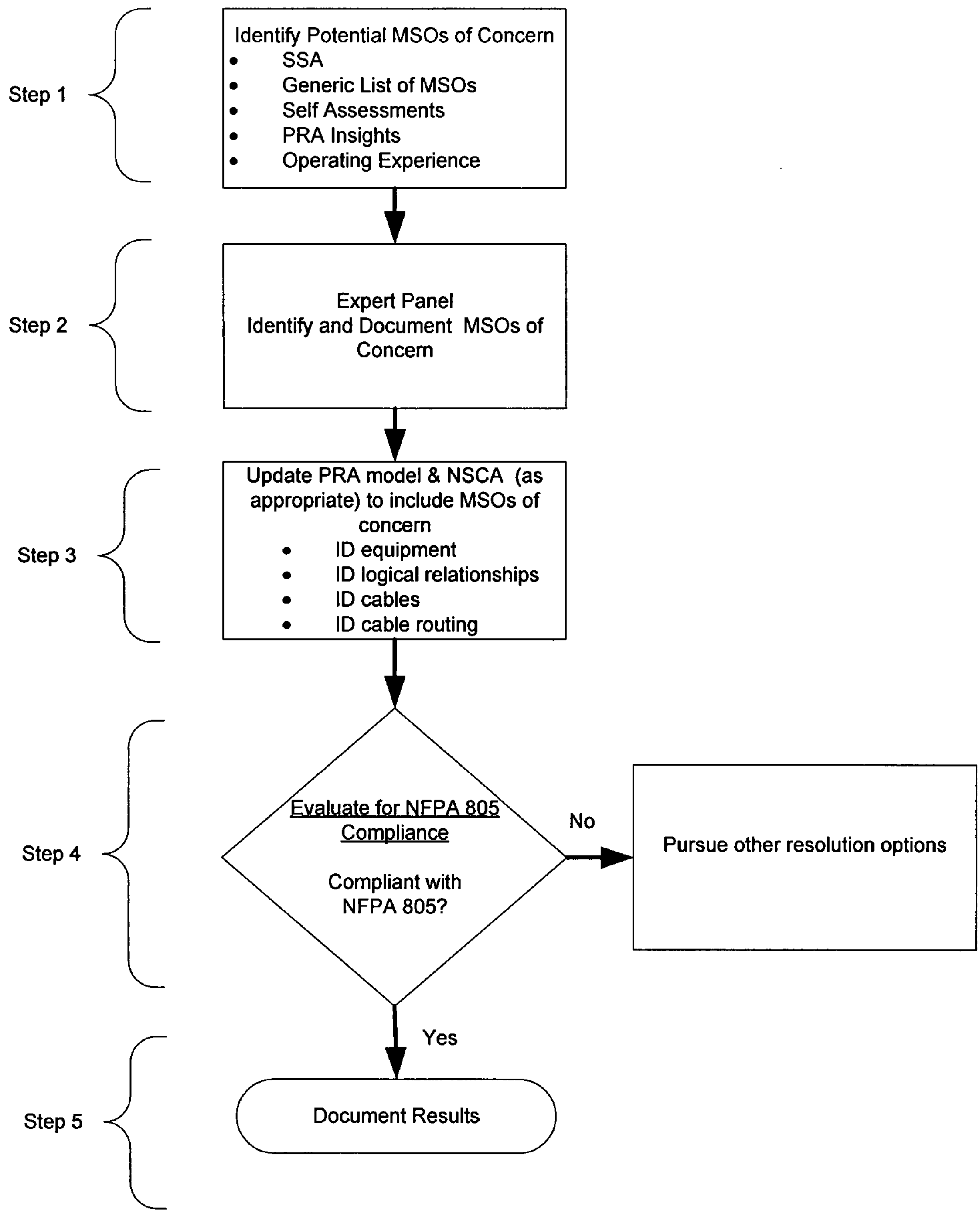

Figure 4-3 – Multiple Spurious Operations – Transition Resolution Process (Based on FAQ 07-0038)

Results

Refer to Attachment F for the process used by BSEP.

4.2.2 Existing Engineering Equivalency Evaluation Transition

Overview of Evaluation Process

The EEEEs that support compliance with NFPA 805 Chapter 3 or Chapter 4 (i.e., both those that existed prior to the transition and those that were created during the transition) were reviewed using the methodology contained in NEI 04-02. The methodology for performing the EEEE review included the following determinations:

- The EEEE is not based solely on quantitative risk evaluations,
- The EEEE is an appropriate use of an engineering equivalency evaluation,
- The EEEE is of appropriate quality,
- The standard license condition is met,
- The EEEE is technically adequate,
- The EEEE reflects the plant as-built condition, and
- The basis for acceptability of the EEEE remains valid

In accordance with the guidance in RG 1.205, Regulatory Position 2.3.2 and NEI 04-02, as clarified by FAQ 07-0054, Demonstrating Compliance with Chapter 4 of NFPA 805, EEEEs that demonstrate that a fire protection system or feature is "adequate for the hazard" are summarized in the LAR as follows:

- If not requesting specific approval for "adequate for the hazard" EEEEs, then the EEEE was referenced where required and a brief description of the evaluated condition was provided.
- If requesting specific NRC approval for "adequate for the hazard" EEEEs, then EEEE was referenced where required to demonstrate compliance and was included in Attachment L for NRC review and approval.

In all cases, the reliance on EEEEs to demonstrate compliance with NFPA 805 requirements was documented in the LAR.

Results

The review results for EEEEs are documented in Attachment A.

In accordance with the guidance provided in RG 1.205, Regulatory Position 2.3.2, NEI 04-02, as clarified by FAQ 07-0054, Demonstrating Compliance with Chapter 4 of NFPA 805, EEEEs used to demonstrate compliance with Chapters 3 and 4 of NFPA 805 are referenced in the Attachments A and C as appropriate.

None of the transitioning EEEEs require NRC approval.

4.2.3 Licensing Action Transition

Overview of Evaluation Process

The existing licensing actions (i.e., Appendix R exemptions) review was performed in accordance with NEI 04-02. The methodology for the licensing action review included the following:

- Determination of the bases for acceptability of the licensing action.
- Determination that these bases for acceptability are still valid and required for NFPA 805.
- Additionally, variances from the deterministic requirements were identified in the NEI 04-02, Table B-3 (See Attachment C). Some of these variances were subsequently dispositioned via the use of the performance-based approach.

Results

Attachment K contains the detailed results of the Licensing Action Review.

None of the licensing actions will be transitioned into the NFPA 805 fire protection program. The licensing actions listed in Attachment K are no longer necessary and will not be transitioned into the NFPA 805 fire protection program. The justifications, grouped by the nature of the exemption, are provided in Attachment O, Orders and Exemptions.

Since the exemptions are either compliant with 10 CFR 50.48(c) or no longer necessary, in accordance with the requirements of 10 CFR 50.48(c)(3)(i), CP&L requests that the exemptions listed in Attachment K be rescinded as part of the LAR process. It is CP&L's understanding that implicit in the superseding of the current license condition, all prior fire protection program Safety Evaluations and commitments will be superseded in their entirety.

4.2.4 Fire Area Transition

Overview of Evaluation Process

The Fire Area Transition (i.e., NEI 04-02 Table B-3) was performed using the methodology contained in NEI 04-02 and FAQ 07-0054. The methodology for performing the Fire Area Transition, depicted in Figure 4-4, is outlined as follows:

Step 1 - Assembled documentation. Gathered industry and plant-specific fire area analyses and licensing basis documents.

Step 2 – Documented fulfillment of nuclear safety performance criteria.

- Assessed accomplishment of nuclear safety performance goals. Documented the method of accomplishment, in summary level form, for the fire area.
- Documented evaluation of effects of fire suppression activities. Documented the evaluation of the effects of fire suppression activities on the ability to achieve the nuclear safety performance criteria.
- Performed licensing action reviews. Performed a review of the licensing aspects of the selected fire area and documented the results of the review. See Section 4.2.3.

- Performed existing engineering equivalency evaluation reviews. Performed a review of existing engineering equivalency evaluations, or created new evaluations, documenting the basis for acceptability. See Section 4.2.2.
- Pre-transition OMA reviews. Performed a review of pre-transition OMAs to determine those actions taking place outside of the main control room or outside of the primary control station(s). See Section 4.2.1.3.

Step 3 – VFDR Identification and characterization and resolution considerations. Identified variances from the deterministic requirements of NFPA 805, Section 4.2.3. Documented variances as either a separation issue or a degraded fire protection system or feature. Developed VFDR problem statements to support resolution.

Step 4 – Performance-Based evaluations (i.e., Fire Modeling or Fire Risk Evaluations) See Section 4.5.2 for additional information.

Step 5 – Final Disposition.

- Documented final disposition of the VFDRs in the fire safety analysis for each area.
- For recovery action compliance strategies, ensured the manual action feasibility analysis of the required recovery actions was completed. Note: If a recovery action cannot meet the feasibility requirements established per NEI 04-02, then alternate means of compliance was considered.
- Documented the post transition NFPA 805 Chapter 4 compliance basis.

Step 6 – Documented required fire protection systems and features. Reviewed the NFPA 805, Section 4.2.3, compliance strategies (i.e., including fire area licensing actions and engineering evaluations) and the NFPA 805, Section 4.2.4, compliance strategies (i.e., including simplifying deterministic assumptions) to determine the scope of fire protection systems and features 'required' by NFPA 805 Chapter 4. The 'required' fire protection systems and features are subject to the applicable requirements of NFPA 805, Chapter 3.

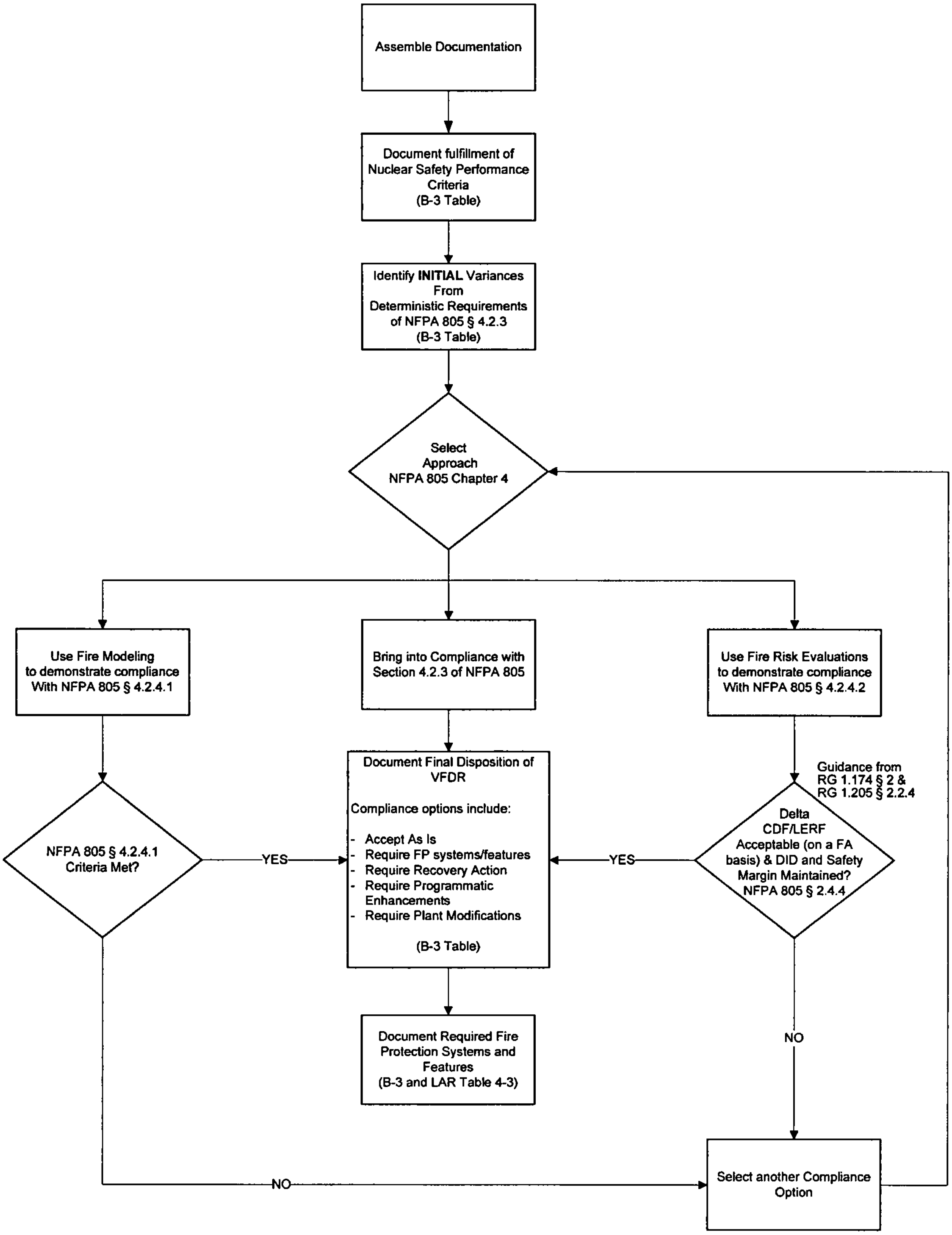

Figure 4-4 – Summary of Fire Area Review [Based on FAQ 07-0054 Revision 1]

Results of the Evaluation Process

Attachment C contains the results of the Fire Area Transition review (i.e., NEI 04-02, Table B-3). On a fire area basis, Attachment C summarizes compliance with Chapter 4 of NFPA 805.

NEI 04-02, Table B-3, includes the following summary level information for each fire area:

- Regulatory Basis – NFPA 805 post-transition regulatory bases are included.
- Performance Goal Summary – An overview of the method of accomplishment of each of the performance criteria in NFPA 805 Section 1.5 is provided.
- Reference Documents – Specific references to Nuclear Safety Capability Assessment Documents are provided.
- Fire Suppression Activities Effect on Nuclear Safety Performance Criteria – A summary of the method of accomplishment is provided.
- Licensing Actions – BSEP is not transitioning any existing Licensing Actions, as noted in Attachment K.
- EEEE – Specific references to EEEE that rely on determinations of "adequate for the hazard" that will remain part of the post-transition licensing basis. A brief description of the condition and the basis for acceptability should be provided.
- VFDRs – Specific variances from the deterministic requirements of NFPA 805 Section 4.2.3. Refer to Section 4.5.2 for a discussion of the performance-based approach.

4.3 Non-Power Operational Modes

4.3.1 Overview of Evaluation Process

BSEP implemented the process outlined in NEI 04-02, Guidance for implementing a Risk-Informed, Performance-Based Program under 10 CFR 50.48(c), and FAQ 07-0040, Clarification on Non-Power Operations. The goal (i.e., as depicted in Figure 4-6) is to ensure that contingency plans are established when the plant is in a Non-Power Operational (NPO) mode where the risk is intrinsically high. During low risk periods, normal risk management controls and fire prevention/protection processes and procedures will be utilized.

The process to demonstrate that the nuclear safety performance criteria are met during NPO modes involved the following steps:

- Reviewed the existing Outage Management Processes
- Identified Equipment/Cables:
    - Reviewed plant systems to determine success paths that support each of the defense-in-depth Key Safety Functions (KSFs), and
    - Identified cables required for the selected components and determined their routing.
- Performed Fire Area Assessments (identify pinch points – plant locations where a single fire may damage all success paths of a KSF).
- Manage pinch-points associated with fire-induced vulnerabilities during the outage.

The process is depicted in Figures 4-5 and 4-6. The results are presented in Section 4.3.2.

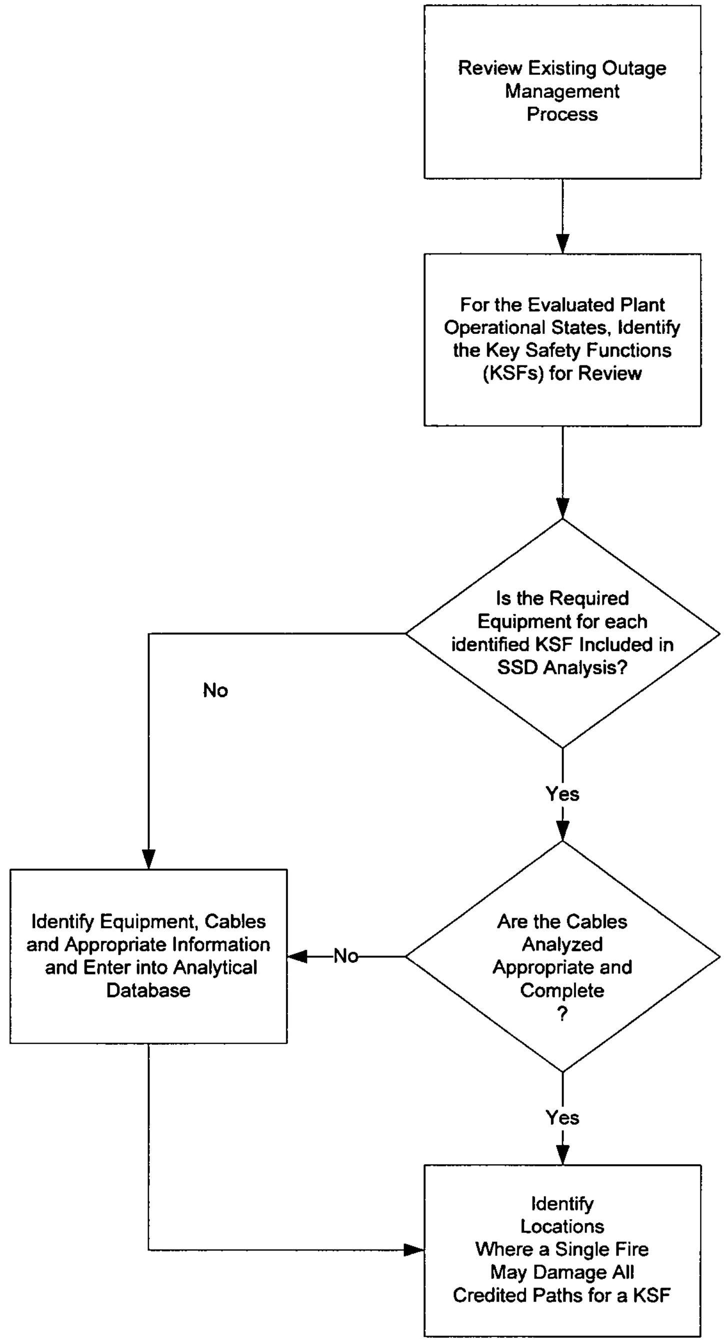

Figure 4-5 Review POSs, KSFs, Equipment, and Cables, and Identify Pinch Points

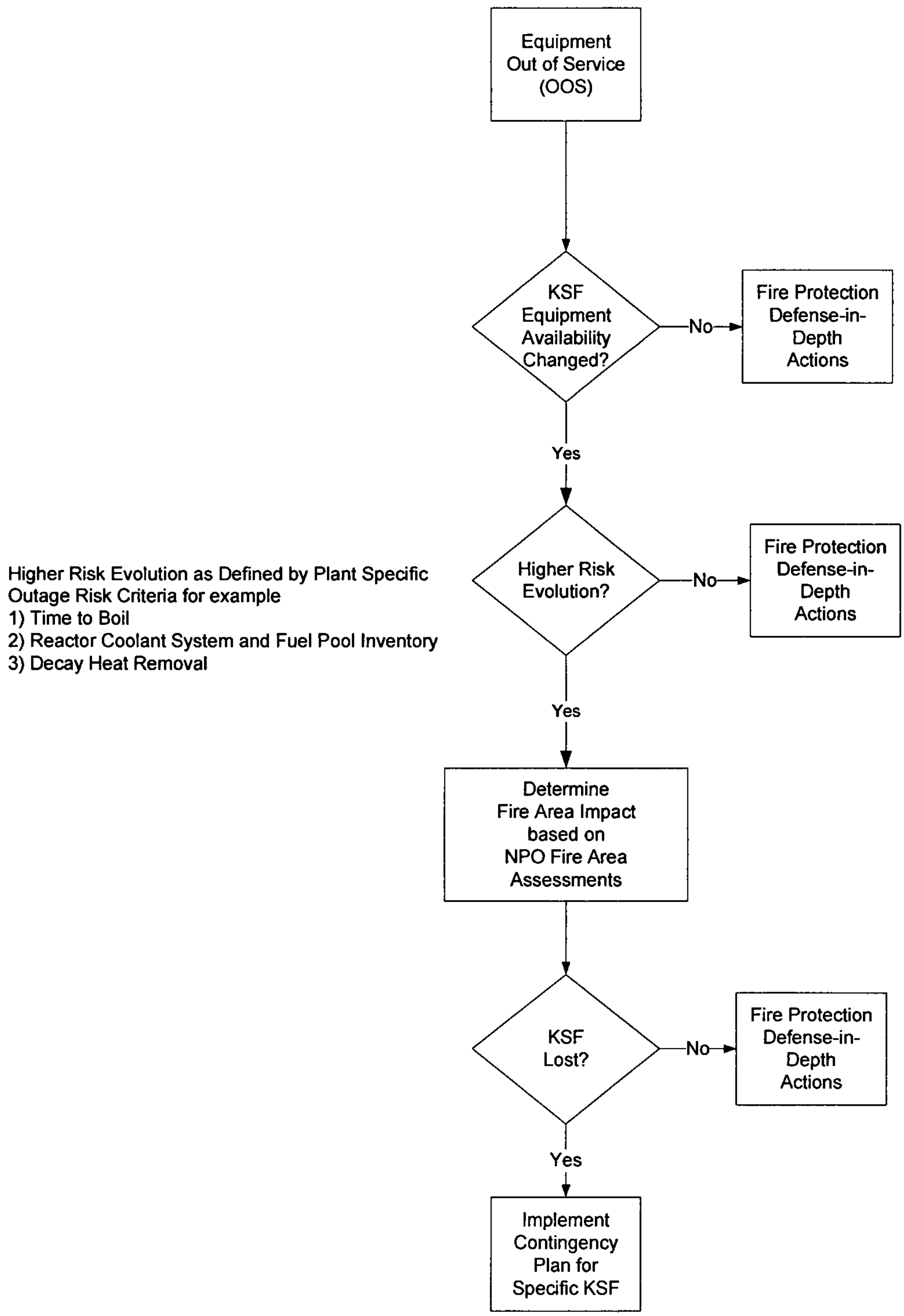

Figure 4-6 Manage Pinch Points

4.3.2 Results of the Evaluation Process

BSEP outage management processes were reviewed. Based on FAQ 07-0040, the Plant Operating States considered for equipment and cable selection are documented in calculation BNP-E-9.011, "NFPA 805 Transition - NPO Modes Review." Using a CAFTA fault tree that models NPO requirements, systems and components were identified to provide three KSFs: Decay Heat Removal, Inventory Control, and Electrical Power Availability (i.e., to the extent that it supports the Decay Heat Removal and Inventory Control functions).

For those components not already in the BSEP Access Database or those with a functional state for non-power operations differing from that in the At-Power Analysis, circuit analysis, cable selection and routing were performed as described in the plant's NSCA methodology. Once all information had been entered into the BSEP Access Database, the ARC™ software package in conjunction with the NPO fault tree was used to determine KSF Pinch Points.

Calculation BNP-E-9.011 provides the results of the fire area assessments for the Pinch Point analysis and provides recommendations for changes to fire risk and outage management procedures and other administrative controls. These include:

- Prohibition or limitation of hot work in fire areas during periods of increased vulnerability.
- Prohibition or limitation of combustible materials in fire areas during periods of increased vulnerability.
- Provision of additional fire watches in affected fire areas during increased vulnerability.
- Identification and monitoring of in-situ ignition sources for "fire precursors" (e.g., equipment temperatures).
- Review of work activities for possible rescheduling
- Equipment realignment (e.g., Swing pumps, Backfeed, etc.)
- Identified procedures to be briefed or walked down.
- Posting of protected equipment.
- Use of recovery actions to mitigate potential losses of KSF success paths.

Attachment D provides a more detailed discussion. Based on incorporation of the recommendations from BNP-E-9.011 into appropriate plant procedures in conjunction with establishment of the NFPA 805 fire protection program, the performance goal for NPO modes (i.e., maintain KSF availability) is fulfilled and the requirements of NFPA 805 are met.

4.4 Radioactive Release Performance Criteria

4.4.1 Overview of Evaluation Process

The review of the fire protection program against NFPA 805 requirements for fire suppression related radioactive release was performed using the methodology contained in NEI 04-02, Table E-1, and was performed using the methodology contained in Project Instruction FPIP-0121, Radiological Release Reviews During Fire Fighting Operations, Rev. 1. The methodology consisted of the following:

- A review of fire pre-plans and fire brigade training materials to identify fire protection program elements (e.g., systems / components / procedural control actions / flow paths) that are being credited to meet the radioactive release goals, objectives, and performance criteria during all plant operating modes, including full power and non-power conditions. Specifically for BSEP, a review was conducted by a review panel to ensure specific steps are included for containment and monitoring of potentially contaminated materials so as to limit the potential for release of radioactive materials due to firefighting operations. The review panel consisted of representatives from Operations, Engineering (i.e., Fire Protection, HVAC Systems), Operations Fire Brigade Training, and Radiation Protection. Site pre-fire plans were screened to identify those locations that have the potential for radiological contamination based on location within plant Radiological Controlled Areas, areas containing potentially contaminated systems, or locations where radioactive materials are routinely stored. In addition, the site fire brigade training materials were reviewed by the same review panel to ensure specific steps are included addressing containment and monitoring of potentially contaminated materials and monitoring of potentially contaminated fire suppression products following a fire event.
- A review of engineering controls to ensure containment of gaseous and liquid effluents (i.e., smoke and fire fighting agents). This review included all plant operating modes (i.e., including full power and non-power conditions). Otherwise, provided a bounding analysis, quantitative analysis, or other analysis that demonstrates that the limitations for instantaneous release of radioactive effluents specified in the unit's Technical Specifications are met.

4.4.2 Results of the Evaluation Process

Fire Pre-Plan review;

The review determined the Fire Protection Program (i.e., Pre-Fire Plans) meets the radioactive release performance criteria by ensuring that radioactive materials (i.e., radiation) generated as a direct result of fire suppression activities is contained and monitored prior to release to unrestricted areas, such that release would be as low as reasonably achievable and would not exceed applicable 10 CFR, Part 20 limits. Containment and monitoring is ensured through elements of the fire brigade training, guidance provided in pre-fire plans and certain plant features (i.e., engineering controls) such as curbs and ventilation systems or actions provided to control smoke management or fire suppression water run-off.

Site specific review of associated fire event and fire suppression related radioactive release is summarized in Attachment E, NEI 04-02, Table E-1. Containment and monitoring actions associated with fire fighting operations are included in the pre-fire plans for fire areas as appropriate based on the screening criteria previously stated (i.e., ref. Table 4-3 and Attachment E) to meet the radiological performance criteria.

The standardized pre-fire plan outline identifies typical fixed radiological hazards for each area. All BSEP pre-fire plans were screened for applicability. Pre-fire plans that address areas where there is no possibility of radiological hazards were screened out from further review. A summary cross-reference of fire compartment, fire area, and pre-fire plan to plant fire areas, and radioactive release input results is provided in Table 4-3. This information was included as input to the individual fire area Fire Safety Analyses (FSA's) calculations. The FSA is the Design Basis Document for NFPA 805 compliance for each fire area and will serve as the location for maintenance and configuration control of the radioactive release review results. Change, modification, or revision to the FSA's is controlled under existing plant engineering configuration control processes.

Table 4-3 -BSEP Pre-Fire Plan Screening

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Unit 1 Cable Access Way (North East Rattle Space) 23ft. Elevation	CB-1	CB-01A	Unit 1 Cable Access Ways	0PFP-CB	1PFP-CB-1	Out	Y
Unit 1 Cable Access Way (North West Rattle Space) 23ft. Elevation	CB-1	CB-01B	Unit 1 Cable Access Ways	0PFP-CB	1PFP-CB-1	Out	Y
Unit 1 Cable Access Way (North East Rattle Space) 49ft. Elevation	CB-1	CB-12A	Unit 1 Cable Access Ways	0PFP-CB	1PFP-CB-12	Out	Y
Unit 1 Cable Access Way (North West Rattle Space) 49ft. Elevation	CB-1	CB-12B	Unit 1 Cable Access Ways	0PFP-CB	1PFP-CB-12	Out	Y
Battery Room 2B 23ft. Elevation	CB-10	CB-10	Unit 2 Division II Battery Room	0PFP-CB	2PFP-CB-10	Out	Y
Unit 2 Cable Access Way (South East Rattle Space) 23ft. Elevation	CB-2	CB-02A	Unit 2 Cable Access Ways	0PFP-CB	2PFP-CB-2	Out	Y
Unit 2 Cable Access Way (South West Rattle Space) 23ft. Elevation	CB-2	CB-02B	Unit 2 Cable Access Ways	0PFP-CB	2PFP-CB-2	Out	Y
Unit 2 Cable Access Way (South East Rattle Space) 49ft. Elevation	CB-2	CB-13A	Unit 2 Cable Access Ways	0PFP-CB	2PFP-CB-13	Out	Y
Unit 2 Cable Access Way (South West Rattle Space) 49ft. Elevation	CB-2	CB-13B	Unit 2 Cable Access Ways	0PFP-CB	2PFP-CB-13	Out	Y
Unit 1 Northwest Stairwell 23ft. and 49ft. Elevations	CB-23E	CB-03	Control Room Extended	0PFP-CB	1PFP-CB-4	Out	Y
Unit 2 Southwest Stairwell 23ft. and 49ft. Elevations	CB-23E	CB-04	Control Room Extended	0PFP-CB	2PFP-CB-3	Out	Y
Unit 1 Cable Spreading Room 23ft. Elevation	CB-23E	CB-05	Control Room Extended	0PFP-CB	1PFP-CB-5	Out	Y
Unit 2 Cable Spreading Room 23ft. Elevation	CB-23E	CB-06	Control Room Extended	0PFP-CB	2PFP-CB-6	Out	Y
Control Building Elevator and Shaft	CB-23E	CB-11	Control Room Extended	0PFP-CB	0PFP-CB-11	Out	Y
Unit 1 Computer Room North 49ft. Elevation	CB-23E	CB-14	Control Room Extended	0PFP-CB	0PFP-CB-14	Out	Y
Unit 2 Computer Room South 49ft. Elevation	CB-23E	CB-15	Control Room Extended	0PFP-CB	0PFP-CB-14	Out	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Auxiliary Operator Briefing Room	CB-23E	CB-16	Control Room Extended	0PFP-CB	0PFP-CB-23	Out	Y
Operator Break Room	CB-23E	CB-17	Control Room Extended	0PFP-CB	0PFP-CB-23	Out	Y
Ladies Washroom	CB-23E	CB-18	Control Room Extended	0PFP-CB	0PFP-CB-23	Out	Y
Central Alarm Station (CAS) 52ft. Elevation	CB-23E	CB-19	Control Room Extended	0PFP-CB	0PFP-CB-19	Out	Y
Unit 1 Northwest Back Panel Area	CB-23E	CB-20	Control Room Extended	0PFP-CB	0PFP-CB-23	Out	Y
Unit 2 Southwest Back Panel Area	CB-23E	CB-21	Control Room Extended	0PFP-CB	0PFP-CB-23	Out	Y
Men's Washroom	CB-23E	CB-22	Control Room Extended	0PFP-CB	0PFP-CB-23	Out	Y
Control Room 49ft. Elevation	CB-23E	CB-23	Control Room Extended	0PFP-CB	0PFP-CB-23	Out	Y
HVAC Equipment Room - 70ft	CB-23E	CB-24	Control Room Extended	0PFP-CB	0PFP-CB-24	Out	Y
Air Conditioning Condenser Area - 70ft	CB-23E	CB-25	Control Room Extended	0PFP-CB	0PFP-CB-25	Out	Y
Control Building Elevator Machinery Room - 70ft	CB-23E	CB-26	Control Room Extended	0PFP-CB	0PFP-CB-26	Out	Y
Battery Room 1A 23ft. Elevation	CB-7	CB-07	Unit 1 Division I Battery Room	0PFP-CB	1PFP-CB-7	Out	Y
Battery Room 1B 23ft. Elevation	CB-8	CB-08	Unit 1 Division II Battery Room	0PFP-CB	1PFP-CB-8	Out	Y
Battery Room 2A 23ft. Elevation	CB-9	CB-09	Unit 2 Division I Battery Room	0PFP-CB	2PFP-CB-9	Out	Y
DG Building Basement - 2ft	DG-1	DG-01	Diesel Generator Basement	0PFP-DG	0PFP-DG-1	Out	N
DG Building Loading Dock, 20ft	DG-10	DG-10	Loading Dock	0PFP-DG	0PFP-013	Out	N
E1 Switchgear Room, 50ft	DG-11	DG-11	E1 Switchgear	0PFP-DG	1PFP-DG-11	Out	N
E2 Switchgear Room, 50ft	DG-12	DG-12	E2 Switchgear	0PFP-DG	1PFP-DG-12	Out	N
E3 Switchgear Room, 50ft	DG-13	DG-13	E3 Switchgear	0PFP-DG	2PFP-DG-13	Out	N
E4 Switchgear Room, 50ft	DG-14	DG-14	E4 Switchgear	0PFP-DG	2PFP-DG-14	Out	N
Supply Air Plenum, 50ft	DG-16E	DG-15	Fan Room Extended	0PFP-DG	0PFP-DG-15	Out	N
Diesel Building Supply Fan Room, 50ft	DG-16E	DG-16	Fan Room Extended	0PFP-DG	0PFP-DG-15	Out	N
Diesel Building North Air Lock, 50ft	DG-16E	DG-17	Fan Room Extended	0PFP-DG	0PFP-DG-15	Out	N

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Diesel Building South Air Lock, 50ft	DG-16E	DG-18	Fan Room Extended	0PFP-DG	0PFP-DG-15	Out	N
AFFF System Room, 50ft	DG-16E	DG-23	Fan Room Extended	0PFP-DG	0PFP-DG-15	Out	N
Diesel Generator Fuel Oil Tank Cell #1, 2ft	DG-19	DG-19	Fuel Oil Tank Cell 1	0PFP-DG	0PFP-DG-19	Out	N
Diesel Generator Cell 4 - 20ft	DG-2	DG-02	Diesel Cell 4	0PFP-DG	2PFP-DG-2	Out	N
Diesel Generator Fuel Oil Tank Cell #2, 2ft	DG-20	DG-20	Fuel Oil Tank Cell 2	0PFP-DG	0PFP-DG-19	Out	N
Diesel Generator Fuel Oil Tank Cell #3, 2ft	DG-21	DG-21	Fuel Oil Tank Cell 3	0PFP-DG	0PFP-DG-19	Out	N
Diesel Generator Fuel Oil Tank Cell #4, 2ft	DG-22	DG-22	Fuel Oil Tank Cell 4	0PFP-DG	0PFP-DG-19	Out	N
Diesel Generator Cell 3 - 20ft	DG-3	DG-03	Diesel Cell 3	0PFP-DG	2PFP-DG-3	Out	N
Diesel Generator Cell 2 - 20ft	DG-4	DG-04	Diesel Cell 2	0PFP-DG	1PFP-DG-4	Out	N
Diesel Generator Cell 1, 20ft	DG-5	DG-05	Diesel Cell 1	0PFP-DG	1PFP-DG-5	Out	N
E5 Switchgear Room, 20ft	DG-6	DG-06	E5 Switchgear	0PFP-DG	1PFP-DG-6	Out	N
E6 Switchgear Room, 20ft	DG-7	DG-07	E6 Switchgear	0PFP-DG	1PFP-DG-7	Out	N
E7 Switchgear Room, 20ft	DG-8	DG-08	E7 Switchgear	0PFP-DG	2PFP-DG-8	Out	N
E8 Switchgear Room, 20ft	DG-9	DG-09	E8 Switchgear	0PFP-DG	2PFP-DG-9	Out	N
Caswell Beach Pumping Station	CASBCH	CASBCH	Caswell Beach Pumping Station	0PFP-MBOCA	0PFP-CAS	Out	N
Hydrogen/Oxygen Storage Facility	HOSF	HOSF	Hydrogen/Oxygen Storage Facility	0PFP-MBOCA	0PFP-HOSF	Out	N
Sodium Hypochlorite Facility	SHF	SHF	Sodium Hypochlorite Facility	0PFP-MBOCA	0PFP-SHF	Out	N
Switchyard	YARD	SY	Yard	0PFP-MBOCA	0PFP-RELAY	Out	N
Auxiliary Boiler House	ABH	ABH	Auxiliary Boiler House	0PFP-MBPA	0PFP-ABH	Out	N
Admin - Annex Building (Security Office Building)	ADANX	ADANX	Admin - Annex Building (Security Office Building)	0PFP-MBPA	0PFP-ADANX	Out	N
Admin Building - First Floor	ADMIN	ADMIN-01	Administration Building	0PFP-MBPA	0PFP-ADMIN-1	Out	N
Admin Building - Second Floor	ADMIN	ADMIN-02	Administration Building	0PFP-MBPA	0PFP-ADMIN-2	Out	N
Clean Maintenance Shop - First Floor	CM	CM-01	Clean Maintenance Shop	0PFP-MBPA	0PFP-CM-1	Out	N

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Clean Maintenance Shop - Second Floor	CM	CM-02	Clean Maintenance Shop	0PFP-MBPA	0PFP-CM-2	Out	N
Fab Shop #1/MOV and Electrical	FAB	FAB-01	Fab Shops	0PFP-MBPA	0PFP-FAB1	Out	N
Fab Shop #3/Diesel Repair	FAB	FAB-03	Fab Shops	0PFP-MBPA	0PFP-FAB3	Out	N
Fire House	FH	FH	Fire House	0PFP-MBPA	0PFP-FH	Out	N
ISFSI Storage Building	ISB	ISB	ISFSI Storage Building	0PFP-MBPA	0PFP-ISFSI-SB	IN	Y
Lube Oil and Paint Storage Building	LUBE	LUBE	Lube Oil and Paint Storage Building	0PFP-MBPA	0PFP-LUBE	Out	N
I&C Breaker Test/NDE Building	MAINT	MAINT	I&C Breaker Test/NDE Building	0PFP-MBPA	0PFP-MAINT	Out	N
Mini Warehouse/Equipment - Outage Storage Building	MINI	MINI	Mini Warehouse/ Equipment - Outage Storage Building	0PFP-MBPA	0PFP-MINI	Out	N
Makeup Water Treatment Building	MWT-1	MWT-01	Makeup Water Treatment	0PFP-MBPA	0PFP-MWT	Out	N
O&M Building - First Floor	OMB	OMB-01	O&M Building	0PFP-MBPA	0PFP-OMB-1	Out	N
O&M Building - Second Floor	OMB	OMB-02	O&M Building	0PFP-MBPA	0PFP-OMB-2	Out	N
O&M Building - Third Floor	OMB	OMB-03	O&M Building	0PFP-MBPA	0PFP-OMB-3	Out	N
Radioactive Material - Container Storage Building	RMCSB	RMCSB	Radioactive Material - Container Storage Building	0PFP-MBPA	0PFP-RMCSB	IN	Y
Secondary Access Point	SAP	SAP	Secondary Access Point	0PFP-MBPA	0PFP-SAP	Out	N
Clean Scaffold Building	SCAFF	SCAFF	Clean Scaffold Building	0PFP-MBPA	0PFP-SCAFF	Out	N
Service Building - First Floor	SERV	SERV-01	Service Building	0PFP-MBPA	0PFP-SERV-1	IN	Y
Service Building - Second Floor	SERV	SERV-02	Service Building	0PFP-MBPA	0PFP-SERV-2	IN	Y
Hot Shop/Stores/Warehouse Building	STORES	STORES	Hot Shop/Stores/ Warehouse Building	0PFP-MBPA	0PFP-STORES	IN	Y
Storm Drain Rad Monitor Building	STORM	STORM	Storm Drain Rad Monitor Building	0PFP-MBPA	0PFP-STORM	IN	Y
Augmented Off-Gas Building	AOG-1	AOG-01	Augmented Off-Gas Building	0PFP-PBAA	0PFP-AOG-1	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Unit 1 Recirc Power Distribution Center (PDC)	RPDC1	RPDC1	Unit 1 Recirc Power Distribution Center (PDC)	0PFP-PBAA	0PFP-RPDC	Out	N
Unit 2 Recirc Power Distribution Center (PDC)	RPDC2	RPDC2	Unit 2 Recirc Power Distribution Center (PDC)	0PFP-PBAA	0PFP-RPDC	Out	N
Radwaste Building Tank Room, minus 3ft	RW-1	RW-01A	Radwaste Building	0PFP-PBAA	0PFP-RW-1a	IN	Y
Radwaste Building CFD Area, 23ft	RW-1	RW-01B	Radwaste Building	0PFP-PBAA	0PFP-RW-1b	IN	Y
Radwaste Building Processing, 35ft	RW-1	RW-01C	Radwaste Building	0PFP-PBAA	0PFP-RW-1c	IN	Y
Radwaste Building Processing, 47ft	RW-1	RW-01D	Radwaste Building	0PFP-PBAA	0PFP-RW-1d	IN	Y
Radwaste Building Roof, 44ft	RW-1	RW-01E	Radwaste Building	0PFP-PBAA	0PFP-RW-1e	IN	Y
Radwaste Building Elevator Machinery Room, 70ft	RW-1	RW-01F	Radwaste Building	0PFP-PBAA	0PFP-RW-1f	IN	Y
Service Water Building Pump Area, 20ft	SW1-1	SW1-01A	Service Water Building	0PFP-PBAA	0PFP-SW-1a	Out	N
Service Water Building Basement, 4ft	SW1-1	SW1-01B	Service Water Building	0PFP-PBAA	0PFP-SW-1b	Out	N
Service Water Building Sump, minus 13ft	SW1-1	SW1-01C	Service Water Building	0PFP-PBAA	0PFP-SW-1b	Out	N
East Yard Open Area	YARD	EY	Yard	0PFP-PBAA	0PFP-EY	IN	N
Transformer Yard	YARD	TY	Yard	0PFP-PBAA	0PFP-TY	Out	N
Chlorination Building	CLB	CLB	Chlorination Building	0PFP-PBAA	0PFP-EY	Out	N
Unit 1 Condensate Transfer Pump House	CTPH1	CTPH1	Unit 1 Condensate Transfer Pump House	0PFP-PBAA	0PFP-EY	IN	Y
Unit 2 Condensate Transfer Pump House	CTPH2	CTPH2	Unit 2 Condensate Transfer Pump House	0PFP-PBAA	0PFP-EY	IN	Y
Duct Bank under East Yard	DUCTBANK	DUCTBANK	DUCTBANK	0PFP-PBAA	0PFP-EY	Out	N
Unit 1 HPCI CO2 Bottle Room	HCB1	HCB1	Unit 1 HPCI CO2 Bottle Room	0PFP-PBAA	0PFP-TY	IN	Y
Unit 2 HPCI CO2 Bottle Room	HCB2	HCB2	Unit 2 HPCI CO2 Bottle Room	0PFP-PBAA	0PFP-TY	IN	Y
HVAC Cooling Towers	HCT	HCT	HVAC Cooling Towers	0PFP-PBAA	0PFP-TY	Out	N
Old NDE Shack	NDE	NDE	Old NDE Shack	0PFP-PBAA	0PFP-EY	Out	N

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Stack Filter House	SFH	SFH	Stack Filter House	0PFP-PBAA	0PFP-EY	IN	Y
Stack Monitoring House	SMH	SMH	Stack Monitoring House	0PFP-PBAA	0PFP-EY	Out	N
Sewage Treatment Plant	STP	STP	Sewage Treatment Plant	0PFP-PBAA	0PFP-TY	Out	N
Unit 1 Valve Pit	VP1	VP1	Unit 1 Valve Pit	0PFP-PBAA	0PFP-EY	Out	N
Unit 2 Valve Pit	VP2	VP2	Unit 2 Valve Pit	0PFP-PBAA	0PFP-EY	Out	N
Circwater Yard	YARD	CW	Yard	0PFP-PBAA	0PFP-EY	Out	N
Radwaste Loading Dock	YARD	RWLD	Yard	0PFP-PBAA	0PFP-RW-1b	IN	Y
Northwest Yard	YARD	NWY	Yard	0PFP-PBAA	0PFP-TY	Out	N
Radwaste Building Elevator	RW-1	RW-01G	Radwaste Building	0PFP-PBAA	0PFP-RW-1f	Out	Y
Reactor Building Southwest Core Spray, minus 17ft	RB1-1	RB1-01A	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1a	IN	Y
Reactor Building Northwest Core Spray, minus 17ft	RB1-1	RB1-01B	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1b	IN	Y
Reactor Building Northeast RHR Room, minus 17ft	RB1-1	RB1-01C	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1c	IN	Y
Reactor Building Southeast RHR Room, minus 17ft	RB1-1	RB1-01D	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1d	IN	Y
Reactor Building Northeast RHR Heat Exchanger, 20ft	RB1-1	RB1-01E	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1e	IN	Y
Reactor Building Southeast RHR Heat Exchanger, 20ft	RB1-1	RB1-01F	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1f	IN	Y
Reactor Building East Central, 20ft	RB1-1	RB1-01G(EC)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1g N 1PFP-RB1-1g S	IN	Y
Reactor Building North Central, 20ft	RB1-1	RB1-01G(NC)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1g N	IN	Y
Reactor Building Northeast Corner, 20ft	RB1-1	RB1-01G(NE)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1g N	IN	Y
Reactor Building Northwest Corner, 20ft	RB1-1	RB1-01G(NW)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1g N	IN	Y
Reactor Building South Central, 20ft	RB1-1	RB1-01G(SC)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1g S	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Reactor Building Southeast Corner, 20ft	RB1-1	RB1-01G(SE)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1g S	IN	Y
Reactor Building Southwest Corner, 20ft	RB1-1	RB1-01G(SW)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1g S	IN	Y
Reactor Building East Central, 50ft	RB1-1	RB1-01H(EC)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1h E	IN	Y
Reactor Building North Central, 50ft	RB1-1	RB1-01H(NC)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1h E 1PFP-RB1-1h W	IN	Y
Reactor Building Northeast Corner, 50ft	RB1-1	RB1-01H(NE)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1h E	IN	Y
Reactor Building Northwest Corner, 50ft	RB1-1	RB1-01H(NW)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1h W	IN	Y
Reactor Building Southeast Corner, 50ft	RB1-1	RB1-01H(SE)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1h E	IN	Y
Reactor Building Southwest Corner, 50ft	RB1-1	RB1-01H(SW)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1h W	IN	Y
Reactor Building West Central, 50ft	RB1-1	RB1-01H(WC)	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1h W	IN	Y
Reactor Building RWCU Access Room, 77ft	RB1-1	RB1-01I	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-12	IN	Y
Reactor Building West, 80ft	RB1-1	RB1-01J	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1j W	IN	Y
Reactor Building East, 80ft	RB1-1	RB1-01K	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1k	IN	Y
Reactor Building Spent Fuel Pool, 117ft	RB1-1	RB1-01L	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1m	IN	Y
Reactor Building Refueling Floor, 117ft	RB1-1	RB1-01M	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1m	IN	Y
Reactor Building HPCI Roof Mezzanine, 5ft	RB1-1	RB1-01N	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1e 1PFP-RB1-1f	IN	Y
Reactor Building ECCS Tunnel Roof, 36ft	RB1-1	RB1-01O	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1o	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Reactor Building HPCI Room, minus 17ft	RB1-1	RB1-02	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-2	IN	Y
Reactor Building Drywell and Torus	RB1-1	RB1-03	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-3	IN	Y
Reactor Building MSIV Pit, 50ft	RB1-1	RB1-04	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-4	IN	Y
Reactor Building HP Field Office, Decontamination Room, 20ft	RB1-1	RB1-05	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1gS	IN	Y
Reactor Building Drywell Entry, 20ft	RB1-1	RB1-07	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-7	IN	Y
Reactor Building TIP Room, 20ft	RB1-1	RB1-08	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-8	IN	Y
Reactor Building Elevator Shaft	RB1-1	RB1-09	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-9	IN	Y
Reactor Building RWCU Pump and Heat Exchanger Room, 50ft	RB1-1	RB1-10	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-10	IN	Y
Reactor Building New Fuel Vault, 117ft	RB1-1	RB1-11	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-16	IN	Y
Reactor Building RWCU Backwash Tank Room, 77ft	RB1-1	RB1-12	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-12	IN	Y
Reactor Building CRD Repair Room, 80ft East	RB1-1	RB1-13	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-13	IN	Y
Reactor Building Skimmer Surge Tank Vault, 117ft	RB1-1	RB1-14	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-14	IN	Y
Reactor Building Elevator Machinery Room, 133ft	RB1-1	RB1-15	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-15	IN	Y
Reactor Building 1A RWCU Filter Pit, 117ft	RB1-1	RB1-16	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-16	IN	Y
Reactor Building 1B RWCU Filter Pit, 117ft	RB1-1	RB1-17	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-16	IN	Y
Reactor Building Supply Room, 98ft	RB1-1	RB1-18	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1p	IN	Y
Reactor Building Platform, 98ft	RB1-1	RB1-18GA	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-1p 1PFP-RB1-19	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Reactor Building Clothing Change room, 98ft	RB1-1	RB1-19	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-19	IN	Y
Reactor Building RWCU Valve Room, 77ft	RB1-1	RB1-20	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-12	IN	Y
Reactor Building Resin Storage Room, 80ft East	RB1-1	RB1-21	Unit 1 Reactor Building General Areas	1PFP-RB	1PFP-RB1-21	IN	Y
Reactor Building ECCS Mini Steam Tunnel, 20ft	RB1-6	RB1-06	Mini Steam Tunnel	1PFP-RB	1PFP-RB1-6	IN	Y
Electrical Tunnel, 9ft	TB1	ET	Turbine Building General Areas	1PFP-TB	1PFP-TB1-ET	IN	Y
Pipe Tunnel, -3ft	TB1	PT	Turbine Building General Areas	1PFP-TB	0PFP-RW-1a	IN	Y
Unit 1 TB Breezeway South, 20ft	TB1	TB1-01A	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1	IN	Y
Unit 1 TB Breezeway North, 20ft	TB1	TB1-01B	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1	IN	Y
Unit 1 TB Mechanical Vacuum Pump Area, 20ft	TB1	TB1-01C	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1c	IN	Y
Unit 1 TB Air Compressor Area, 20ft	TB1	TB1-01D	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1d	IN	Y
Unit 1 TB 2A Air Dryer Area, 20ft	TB1	TB1-01E	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1e	IN	Y
Unit 1 TB 4KV Switchgear Area, 20ft	TB1	TB1-01F	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1f	IN	Y
Unit 1 TB Hydrogen Seal Oil Area, 20ft	TB1	TB1-01G	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1g	IN	Y
Unit 1 TB Condensate Pump Area, 20ft	TB1	TB1-01H	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1h	IN	Y
Unit 1 TB 1A Reactor Recirc MG Set Room, 38ft	TB1	TB1-01I	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1i & j	IN	Y
Unit 1 TB 1B Reactor Recirc MG Set Room, 38ft	TB1	TB1-01J	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1i & j	IN	Y
Unit 1 TB South 38ft and 45ft	TB1	TB1-01K	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1k	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Unit 1 TB Supply Fan Room 55ft	TB1	TB1-01L	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1l	IN	Y
Unit 1 TB Main Turbine Front Standard Area, 70ft	TB1	TB1-01M	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1m	IN	Y
Unit 1 TB Main Turbine and MSR Area, 70ft	TB1	TB1-01N	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1n	IN	Y
Unit 1 TB Main Generator and Exciter area, 70ft	TB1	TB1-01O	Turbine Building General Areas	1PFP-TB	1PFP-TB1-1o	IN	Y
Unit 1 TB 1B SJAE Room, 20ft	TB1	TB1-02	Turbine Building General Areas	1PFP-TB	1PFP-TB1-2	IN	Y
Unit 1 TB 1A SJAE Room, 20ft	TB1	TB1-03	Turbine Building General Areas	1PFP-TB	1PFP-TB1-3	IN	Y
Unit 1 TB 1B RFPT Room, 20ft	TB1	TB1-04	Turbine Building General Areas	1PFP-TB	1PFP-TB1-4	IN	Y
Unit 1 TB 1A RFPT Room, 20ft	TB1	TB1-05	Turbine Building General Areas	1PFP-TB	1PFP-TB1-5	IN	Y
Unit 1 TB Condensate Booster Pump Room, 20ft	TB1	TB1-06	Turbine Building General Areas	1PFP-TB	1PFP-TB1-6	IN	Y
Unit 1 TB Heater Drain Pump Room, 9ft	TB1	TB1-07	Turbine Building General Areas	1PFP-TB	1PFP-TB1-7	IN	Y
Unit 1 TB Condenser Bay Area, 20ft	TB1	TB1-08A	Turbine Building General Areas	1PFP-TB	1PFP-TB1-8a	IN	Y
Unit 1 TB Condenser Pit East Area, 20ft	TB1	TB1-08B	Turbine Building General Areas	1PFP-TB	1PFP-TB1-8b	IN	Y
Unit 1 TB Condenser Pit West Area, 20ft	TB1	TB1-08C	Turbine Building General Areas	1PFP-TB	1PFP-TB1-8c	IN	Y
Unit 1 TB Condenser Bay Area, 45ft	TB1	TB1-08D	Turbine Building General Areas	1PFP-TB	1PFP-TB1-8d	IN	Y
Unit 1 TB Condenser Pit East Area, 45ft	TB1	TB1-08E	Turbine Building General Areas	1PFP-TB	1PFP-TB1-8e	IN	Y
Unit 1 TB Condenser Pit West Area, 45ft	TB1	TB1-08F	Turbine Building General Areas	1PFP-TB	1PFP-TB1-8f	IN	Y
Unit 1 TB EHC and Lube Oil Room, 20ft	TB1	TB1-09A	Turbine Building General Areas	1PFP-TB	1PFP-TB1-9a	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Unit 1 TB EHC and Lube Oil Room, 45ft	TB1	TB1-09B	Turbine Building General Areas	1PFP-TB	1PFP-TB1-9b	IN	Y
Unit 1 TB Exhaust Fan Room, 45ft	TB1	TB1-10	Turbine Building General Areas	1PFP-TB	1PFP-TB1-10	IN	Y
Unit 1 TB A Train HP Feedwater heater Room, 45ft	TB1	TB1-12	Turbine Building General Areas	1PFP-TB	1PFP-TB1-12	IN	Y
Unit 1 TB B Train HP Feedwater heater Room, 45ft	TB1	TB1-13	Turbine Building General Areas	1PFP-TB	1PFP-TB1-13	IN	Y
Unit 1 TB Receiving Area	TB1	TB1-15	Turbine Building General Areas	1PFP-TB	0PFP-013	IN	Y
Unit 1 Heater Bay Roof	YARD	TB1-HBROOF	Yard	1PFP-TB	1PFP-TB1-1m	IN	Y
Turbine Building 1 Dragon's Breath	TB1	TB1-DB	Turbine Building General Areas	1PFP-TB	1PFP-TB1-12	IN	Y
Reactor Building Southwest Core Spray, minus 17ft	RB2-1	RB2-01A	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1a	IN	Y
Reactor Building Northwest Core Spray, minus 17ft	RB2-1	RB2-01B	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1b	IN	Y
Reactor Building Northeast RHR Room, minus 17ft	RB2-1	RB2-01C	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1c	IN	Y
Reactor Building Southeast RHR Room, minus 17ft	RB2-1	RB2-01D	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1d	IN	Y
Reactor Building Northeast RHR Heat Exchanger, 20ft	RB2-1	RB2-01E	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1e	IN	Y
Reactor Building Southeast RHR Heat Exchanger, 20ft	RB2-1	RB2-01F	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1f	IN	Y
Reactor Building East Central, 20ft	RB2-1	RB2-01G(EC)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1g N 2PFP-RB2-1g S	IN	Y
Reactor Building North Central, 20ft	RB2-1	RB2-01G(NC)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1g N	IN	Y
Reactor Building Northeast Corner, 20ft	RB2-1	RB2-01G(NE)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1g N	IN	Y
Reactor Building Northwest Corner, 20ft	RB2-1	RB2-01G(NW)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1g N	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Reactor Building South Central, 20ft	RB2-1	RB2-01G(SC)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1g S	IN	Y
Reactor Building Southeast Corner, 20ft	RB2-1	RB2-01G(SE)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1g S	IN	Y
Reactor Building Southwest Corner, 20ft	RB2-1	RB2-01G(SW)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1g S	IN	Y
Reactor Building East Central, 50ft	RB2-1	RB2-01H(EC)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1h E	IN	Y
Reactor Building North Central, 50ft	RB2-1	RB2-01H(NC)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1h E 2PFP-RB2-1h W	IN	Y
Reactor Building Northeast Corner, 50ft	RB2-1	RB2-01H(NE)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1h E	IN	Y
Reactor Building Northwest Corner, 50ft	RB2-1	RB2-01H(NW)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1h W	IN	Y
Reactor Building Southeast Corner, 50ft	RB2-1	RB2-01H(SE)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1h E	IN	Y
Reactor Building Southwest Corner, 50ft	RB2-1	RB2-01H(SW)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1h W	IN	Y
Reactor Building West Central, 50ft	RB2-1	RB2-01H(WC)	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1h W	IN	Y
Reactor Building RWCU Access Room, 77ft	RB2-1	RB2-01I	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-12	IN	Y
Reactor Building West, 80ft	RB2-1	RB2-01J	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1j W	IN	Y
Reactor Building East, 80ft	RB2-1	RB2-01K	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1k	IN	Y
Reactor Building Spent Fuel Pool, 117ft	RB2-1	RB2-01L	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1m	IN	Y
Reactor Building Refueling Floor, 117ft	RB2-1	RB2-01M	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1m	IN	Y
Reactor Building HPCI Roof Mezzanine, 5ft	RB2-1	RB2-01N	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1e 2PFP-RB2-1f	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Reactor Building ECCS Tunnel Roof, 36ft	RB2-1	RB2-01O	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1o	IN	Y
Reactor Building HPCI Room, minus 17ft	RB2-1	RB2-02	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-2	IN	Y
Reactor Building Drywell and Torus	RB2-1	RB2-03	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-3	IN	Y
Reactor Building MSIV Pit, 50ft	RB2-1	RB2-04	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-4	IN	Y
Reactor Building HP Field Office, Decontamination Room, 20ft	RB2-1	RB2-05	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1g	IN	Y
Reactor Building Drywell Entry, 20ft	RB2-1	RB2-07	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-7	IN	Y
Reactor Building TIP Room, 20ft	RB2-1	RB2-08	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-8	IN	Y
Reactor Building Elevator Shaft	RB2-1	RB2-09	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-9	IN	Y
Reactor Building RWCU Pump and Heat Exchanger Room, 50ft	RB2-1	RB2-10	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-10	IN	Y
Reactor Building New Fuel Vault, 117ft	RB2-1	RB2-11	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1m	IN	Y
Reactor Building RWCU Backwash Tank Room, 77ft	RB2-1	RB2-12	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-12	IN	Y
Reactor Building CRD Repair Room, 80ft East	RB2-1	RB2-13	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-13	IN	Y
Reactor Building Skimmer Surge Tank Vault, 117ft	RB2-1	RB2-14	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-14	IN	Y
Reactor Building Elevator Machinery Room, 133ft	RB2-1	RB2-15	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-15	IN	Y
Reactor Building 2A RWCU Filter Pit, 117ft	RB2-1	RB2-16	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-16	IN	Y
Reactor Building 2B RWCU Filter Pit, 117ft	RB2-1	RB2-17	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-16	IN	Y
Reactor Building Supply Room, 98ft	RB2-1	RB2-18	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1p	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Reactor Building Platform, 98ft	RB2-1	RB2-18GA	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-1p	IN	Y
Reactor Building Clothing Change room, 98ft	RB2-1	RB2-19	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-19	IN	Y
Reactor Building Resin Storage Room, 80ft East	RB2-1	RB2-20	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-20	IN	Y
Reactor Building RWCU Valve Room, 77ft	RB2-1	RB2-21	Unit 2 Reactor Building General Areas	2PFP-RB	2PFP-RB2-12	IN	Y
Reactor Building ECCS Mini Steam Tunnel, 20ft	RB2-6	RB2-06	Mini Steam Tunnel	2PFP-RB	2PFP-RB2-6	IN	Y
Unit 2 TB Breezeway North	TB1	TB2-01A	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1	IN	Y
Unit 2 TB Breezeway South	TB1	TB2-01B	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1	IN	Y
Unit 2 TB Mechanical Vacuum Pump Area, 20ft	TB1	TB2-01C	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1c	IN	Y
Unit 2 TB Air Compressor Area, 20ft	TB1	TB2-01D	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1d	IN	Y
Unit 2 TB 2A Air Dryer Area, 20ft	TB1	TB2-01E	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1e	IN	Y
Unit 2 TB 4KV Switchgear Area, 20ft	TB1	TB2-01F	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1f	IN	Y
Unit 2 TB Hydrogen Seal Oil Area, 20ft	TB1	TB2-01G	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1g	IN	Y
Unit 2 TB Condensate Pump Area, 20ft	TB1	TB2-01H	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1h	IN	Y
Unit 2 TB 2B Reactor Recirc MG Set Room, 38ft	TB1	TB2-01I	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1i & j	IN	Y
Unit 2 TB 2A Reactor Recirc MG Set Room, 38ft	TB1	TB2-01J	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1i & j	IN	Y
Unit 2 TB North 38ft and 45ft	TB1	TB2-01K	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1k	IN	Y
Unit 2 TB Supply Fan Room 55ft	TB1	TB2-01L	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1l	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Unit 2 TB Main Turbine Front Standard Area, 70ft	TB1	TB2-01M	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1m	IN	Y
Unit 2 TB Main Turbine and MSR Area, 70ft	TB1	TB2-01N	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1n	IN	Y
Unit 2 TB Main Generator and Exciter area, 70ft	TB1	TB2-01O	Turbine Building General Areas	2PFP-TB	2PFP-TB2-1o	IN	Y
Unit 2 TB 2B SJAE Room, 20ft	TB1	TB2-02	Turbine Building General Areas	2PFP-TB	2PFP-TB2-2	IN	Y
Unit 2 TB 2A SJAE Room, 20ft	TB1	TB2-03	Turbine Building General Areas	2PFP-TB	2PFP-TB2-3	IN	Y
Unit 2 TB 2B RFP Room, 20ft	TB1	TB2-04	Turbine Building General Areas	2PFP-TB	2PFP-TB2-4	IN	Y
Unit 2 TB 2A RFP Room, 20ft	TB1	TB2-05	Turbine Building General Areas	2PFP-TB	2PFP-TB2-5	IN	Y
Unit 2 TB Condensate Booster Pump Room, 20ft	TB1	TB2-06	Turbine Building General Areas	2PFP-TB	2PFP-TB2-6	IN	Y
Unit 2 TB Heater Drain Pump Room, 20ft	TB1	TB2-07	Turbine Building General Areas	2PFP-TB	2PFP-TB2-7	IN	Y
Unit 2 TB Condenser Bay Area, 20ft	TB1	TB2-08A	Turbine Building General Areas	2PFP-TB	2PFP-TB2-8a	IN	Y
Unit 2 TB Condenser Pit East Area, 20ft	TB1	TB2-08B	Turbine Building General Areas	2PFP-TB	2PFP-TB2-8b	IN	Y
Unit 2 TB Condenser Pit West Area, 20ft	TB1	TB2-08C	Turbine Building General Areas	2PFP-TB	2PFP-TB2-8c	IN	Y
Unit 2 TB Condenser Bay Area, 45ft	TB1	TB2-08D	Turbine Building General Areas	2PFP-TB	2PFP-TB2-8d	IN	Y
Unit 2 TB Condenser Pit East Area, 45ft	TB1	TB2-08E	Turbine Building General Areas	2PFP-TB	2PFP-TB2-8e	IN	Y
Unit 2 TB Condenser Pit West Area, 45ft	TB1	TB2-08F	Turbine Building General Areas	2PFP-TB	2PFP-TB2-8f	IN	Y
Unit 2 TB EHC and Lube Oil Room, 20ft	TB1	TB2-09A	Turbine Building General Areas	2PFP-TB	2PFP-TB2-9a	IN	Y
Unit 2 TB EHC and Lube Oil Room, 45ft	TB1	TB2-09B	Turbine Building General Areas	2PFP-TB	2PFP-TB2-9b	IN	Y

Fire Zone Description	Fire Area	Fire Zone	Fire Area Description	Fire Pre-Plan Procedure	Fire Pre-Plan	Rad Release Input (Screened)	In RCA? Y/N
Unit 2 TB Exhaust Fan Room, 45ft	TB1	TB2-10	Turbine Building General Areas	2PFP-TB	2PFP-TB2-10	IN	Y
Unit 2 TB A Train HP Feedwater heater Room, 45ft	TB1	TB2-12	Turbine Building General Areas	2PFP-TB	2PFP-TB2-12	IN	Y
Unit 2 TB B Train HP Feedwater heater Room, 45ft	TB1	TB2-13	Turbine Building General Areas	2PFP-TB	2PFP-TB2-13	IN	Y
Turbine Building Elevator Shaft	TB1	TB2-14	Turbine Building General Areas	2PFP-TB	2PFP-TB2-14	IN	Y
Turbine Building Elevator Machinery Room	TB1	TB2-15	Turbine Building General Areas	2PFP-TB	2PFP-TB2-15	IN	Y
Turbine Building Laydown Area	TB1	TB2-16	Turbine Building General Areas	2PFP-TB	2PFP-TB2-16	IN	Y
Unit 2 Heater Bay Roof	YARD	TB2-HBROOF	Yard	2PFP-TB	2PFP-TB2-1m	IN	Y
Turbine Building 2 Dragon's Breath	TB1	TB2-DB	Turbine Building General Areas	2PFP-TB	2PFP-TB2-12	IN	Y
Unit 1 Control Building Roof	YARD	CB-ROOF1	Yard	0PFP-013	*New PFP	IN	Y
Unit 2 Control Building Roof	YARD	CB-ROOF2	Yard	0PFP-013	*New PFP	IN	Y
Reactor Building 1 Roof	YARD	RB1-ROOF	Yard	0PFP-013	*New PFP	IN	Y

Fire Brigade Training Plan Review;

BSEP has completed transition of its fire brigade and site incident commander lesson plans to a fleet standard, NFPA 600 compliant format, aligning with NFPA 805, Section 3.4.1. Attributes are included within the new NFPA 600 lesson plans to address the Radioactive Release objectives. Lesson plan topics are technical skill-set based rather than fire area specific. As such, discussion points were noted for the topics applicable to, or having potential impact to radioactive release due to firefighting activities. Discussion points are included regarding containment and monitoring of potentially contaminated fire suppression agents and products of combustion for the following lesson plan topical areas;

- Safety and Orientation
- Personnel Protective Equipment
- Fire Hose
- Forcible Entry
- Ventilation
- Overhaul
- Fire Attack

Engineering Controls Review;

The review panel determined Engineering Controls are adequate to ensure that radioactive materials (i.e., radiation) generated as a direct result of fire suppression activities is contained and monitored prior to release to unrestricted areas such that such release would be as low as reasonably achievable and would not exceed applicable 10 CFR, Part 20 limits. Engineering controls such as use of forced air ventilation and damming for fire suppression agent run-off was considered during review of fire pre-plans, for areas in which this is the anticipated response identified in the pre-fire plan. No new engineering controls were identified or established as a result of this review, and all present controls are as currently in place under the approved pre-transitional fire protection program.

Documentation;

Results of the radioactive release reviews described above have been documented in summary format in Attachment E. Open Items identified in the review process will be incorporated into the indicated fire pre-plans.

4.5 Fire PRA and Performance-Based Approaches

RI-PB evaluations are an integral element of an NFPA 805 fire protection program. Key parts of RI-PB evaluations include:

- A Fire PRA (i.e., discussed in Section 4.5.1 and Attachments U, V, and W).
- NFPA 805 Performance-Based Approaches (i.e., discussed in Section 4.5.2).

4.5.1 Fire PRA Development and Assessment

In accordance with the guidance in RG 1.205, a Fire PRA model was developed for BSEP in compliance with the requirements of Part 4, "Requirements for Fires At Power PRA," of the ASME and ANS combined PRA Standard, ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Application," (i.e., hereafter referred to as Fire PRA Standard). CP&L conducted a peer review by independent industry analysts in accordance with RG 1.200 prior to a risk-informed submittal. The resulting fire risk assessment model is used as the analytical tool to perform Fire Risk Evaluations during the transition process.

Section 4.5.1.1 describes the Internal Events PRA model. Section 4.5.1.2 describes the Fire PRA model. Section 4.5.1.3 describes the results and resolution of the peer review of the Fire PRA, and Section 4.5.1.4 describes insights gained from the Fire PRA.

4.5.1.1 Internal Events PRA

The Brunswick Unit 1 and 2 base internal events PRA (i.e., Calculation BNP-PSA-030) was the starting point for the Fire PRA. Attachment U provides a discussion of the internal events PRA and the results and disposition of the most recent peer review.

4.5.1.2 Fire PRA

The internal events PRA was modified to capture the effects of fire both as an initiator of an event and as a potential failure mode of affected circuits and individual targets. The Fire PRA was developed using the guidance for Fire PRA development in NUREG/CR-6850/EPRI TR 1011989, approved FAQs, and EPRI TR 1016735.

The Fire PRA quality and results are discussed in the subsequent sections and in Attachments V and W, respectively.

Fire Model Utilization in the Application

RG 1.205, Regulatory Position 4.2 and Section 5.1.2 of NEI 04-02, provide guidance on documenting the fire models used, and justifying that these fire models and methods are acceptable for use in performance-based analyses when performed by qualified users, have been verified and validated, and are used within their limitations and with the rigor required by the nature and scope of the analyses.

As part of the NFPA 805 transition, fire modeling was performed as part of the Fire PRA development (i.e., NFPA 805 Section 4.2.4.2) and, therefore, maximum expected fire scenario (MEFS)/limiting fire scenario (LFS) were not analyzed separately. RG 1.205, Regulatory Position 4.2 and Section 5.1.2 of NEI 04-02, provide guidance to identify fire models that are acceptable to the NRC for plants implementing a risk-informed, performance-based licensing basis.

The following fire models were used:

- Fire Dynamics Tools (FDT's)
- Consolidated Model of Fire and Smoke Transport (CFAST)
- Fire Dynamics Simulator (FDS)

The approach taken at BSEP to simplify the analysis process incorporates features of several fire model tools covered by NUREG-1824, as well as additional features. The approach is collectively referred to as the Fire Modeling Generic Treatments. The analysis basis and Verification and Validation (V&V) documentation was provided in a proprietary Hughes Associates, Inc. report to the NRC on January 24, 2008. The report entitled "Generic Fire Modeling Treatments" is effectively a technical reference guide, a user's guide, and the V&V basis.

The use of the Generic Treatments in specific applications at BSEP falls within their limitations as described in the "Generic Fire Modeling Treatments". In addition to the generic fire modeling treatments that were used in the hazard analysis, several calculations were produced that used FDS, CFAST, and the FDT's as documented in NUREG-1824.

The acceptability of the use of these fire models is included in Attachment J.

4.5.1.3 Results of Fire PRA Peer Review

The Brunswick Unit 1 and 2 Fire PRA (i.e., Calculations BNP-PSA-080 and BNP-PSA-082) was peer reviewed against the requirements of ASME/ANS RA-Sa-2009, Part 4 and Regulatory Guide 1.200, revision 2.

The results (i.e., Supporting Requirement capability assessments and Facts & Observations (F&Os)) documented in the February 2012 Fire PRA peer review report, and subsequent focused scope peer review reports, were used to support the Fire PRA for the NFPA 805 application.

The Fire PRA update addressed the Supporting Requirement assessed deficiencies (i.e., Not Met or Capability Category I (CC I)). Completion of recommendations related to Supporting Requirement assessments and 'Finding' F&Os results in a Capability Category II assessment for the associated Supporting Requirements. Some items are not completed at this time and are deferred. These items have been dispositioned for the potential impact on the Fire PRA and the application. The results of the peer review are summarized in Attachment V.

4.5.1.4 Risk Insights

Risk insights were documented as part of the development of the Fire PRA. The total plant fire CDF/LERF was derived using the NUREG/CR-6850 methodology for fire PRA development and is useful in identifying the areas of the plant where fire risk is greatest. A review of the fire initiating events that collectively represent 95% of the calculated fire risk is included as Attachment W.

4.5.2 Performance-Based Approaches

NFPA 805 outlines the approaches for performing performance-based analyses. As specified in Section 4.2.4, there are generally two types of analyses performed for the performance-based approach:

- Fire Modeling (i.e., NFPA 805, Section 4.2.4.1).
- Fire Risk Evaluation (i.e., NFPA 805, Section 4.2.4.2).

4.5.2.1 Fire Modeling Approach

The fire modeling approach was not utilized for demonstrating compliance with NFPA 805 for BSEP.

4.5.2.2 Fire Risk Approach

Overview of Evaluation Process

The Fire Risk Evaluations were completed as part of the BSEP NFPA 805 transition. These Fire Risk Evaluations were developed using the process described below. This methodology is based upon the requirements of NFPA 805, industry guidance in NEI 04-02, and RG 1.205. These are summarized in Table 4-1.

Table 4-1 Fire Risk Evaluation Guidance Summary Table

Document	Section(s)	Topic
NFPA 805	2.2(h), 4.2.4, A.2.2(h), A.2.4.4, D.5	Change Evaluation (2.2(h), 2.2.9, 2.4.4 A.2.2(h), A.2.4.4, D.5) Risk of Recovery Actions (4.2.4) Use of Fire Risk Evaluation (4.2.4.2)
NEI 04-02 Revision 2	4.4, 5.3, Appendix B, Appendix I, Appendix J	Change Evaluation, Change Evaluation Forms (App. I), No specific discussion of Fire Risk Evaluation
RG 1.205 Revision 1	C.2.2.4, C.2.4, C.3.2	Risk Evaluations (C.2.2.4) Recovery Actions (C.2.4)

During the transition to NFPA 805, variances from the deterministic approach in Section 4.2.3 of NFPA 805 were evaluated using a Fire Risk Evaluation per Section 4.2.4.2 of NFPA 805. A Fire Safety Analysis was performed for each fire area. For areas containing variances from the deterministic requirements (VFDRs) of Section 4.2.3 of NFPA 805, a Fire Risk Evaluation was performed for each fire area containing VFDRs.

If the Fire Risk Evaluation meets the acceptance criteria, this is confirmation that a success path effectively remains free of fire damage and that the performance-based approach is acceptable per Section 4.2.4.2 of NFPA 805.

The Fire Risk Evaluation process consists of the following steps (Figure 4-7 depicts the Fire Risk Evaluation process used during transition. This is generally based on FAQ 07-0054 Revision 1:

Step 1 – Preparation for the Fire Risk Evaluation.

- Definition of the Variances from the Deterministic Requirements. The definition of the VFDR includes a description of problem statement and the section of NFPA 805 that is not met, type of VFDR (e.g., separation issue or degraded fire protection system), and proposed evaluation per applicable NFPA 805 section.
- Preparatory Evaluation – Fire Risk Evaluation Team Review. Using the information obtained during the development of the NEI 04-02 B-3 Table and the Fire PRA, a team review of the VFDR was performed. Depending on the scope and complexity of the VFDR, the team may include the Safe shutdown/NSCA

Engineer, the Fire Protection Engineer, and the Fire PRA Engineer. The purpose and objective of this team review was to address the following;

- o Review of the Fire PRA modeling treatment of VFDR
- o Ensure discrepancies were captured and resolved

Step 2 – Performed the Fire Risk Evaluation

- The Evaluator coordinated as necessary with the Safe shutdown/NSCA Engineer, Fire Protection Engineer and Fire PRA Engineer to assess the VFDR using the Fire Risk Evaluation process to perform the following:
  - o Change in Risk Calculation with consideration for additional risk of recovery actions and required fire protection systems and features due to fire risk.
  - o Fire area change in risk summary

Step 3 – Reviewed the Acceptance Criteria

- The acceptance criteria for the Fire Risk Evaluation consist of two parts. One is quantitatively based and the other is qualitatively based. The quantitative figures of merit are ΔCDF and ΔLERF. The qualitative factors are defense-in-depth and safety margin.
  - o Risk Acceptance Criteria. The transition risk evaluation was measured quantitatively for acceptability using the ΔCDF and ΔLERF criteria from RG 1.174, as clarified in RG 1.205, Regulatory Position 2.2.4.
  - o Defense-in-Depth. A review of the impact of the change on defense-in-depth was performed, using the guidance NEI 04-02. NFPA 805 defines defense-in-depth as:
    - Preventing fires from starting
    - Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting damage
    - Providing adequate level of fire protection for structures, systems and components important to safety; so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

    In general, the defense-in-depth requirement was considered to be satisfied if the proposed change does not result in a substantial imbalance among these elements (or echelons).

    The review of defense-in-depth was qualitative and addressed each of the elements with respect to the proposed change. Defense-in-depth was performed on a fire area basis.

    Fire protection features and systems relied upon to ensure defense-in-depth were identified as a result of the assessment of defense-in-depth.
  - o Safety Margin Assessment. A review of the impact of the change on safety margin was performed. An acceptable set of guidelines for making that assessment is summarized below. Other equivalent acceptance guidelines may also be used.

- Codes and standards or their alternatives accepted for use by the NRC are met, and
- Safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met, or provides sufficient margin to account for analysis and data uncertainty.

The requirements related to safety margins for the change analysis are described for each of the specific analysis types used in support of the Fire Risk Evaluation (FRE).

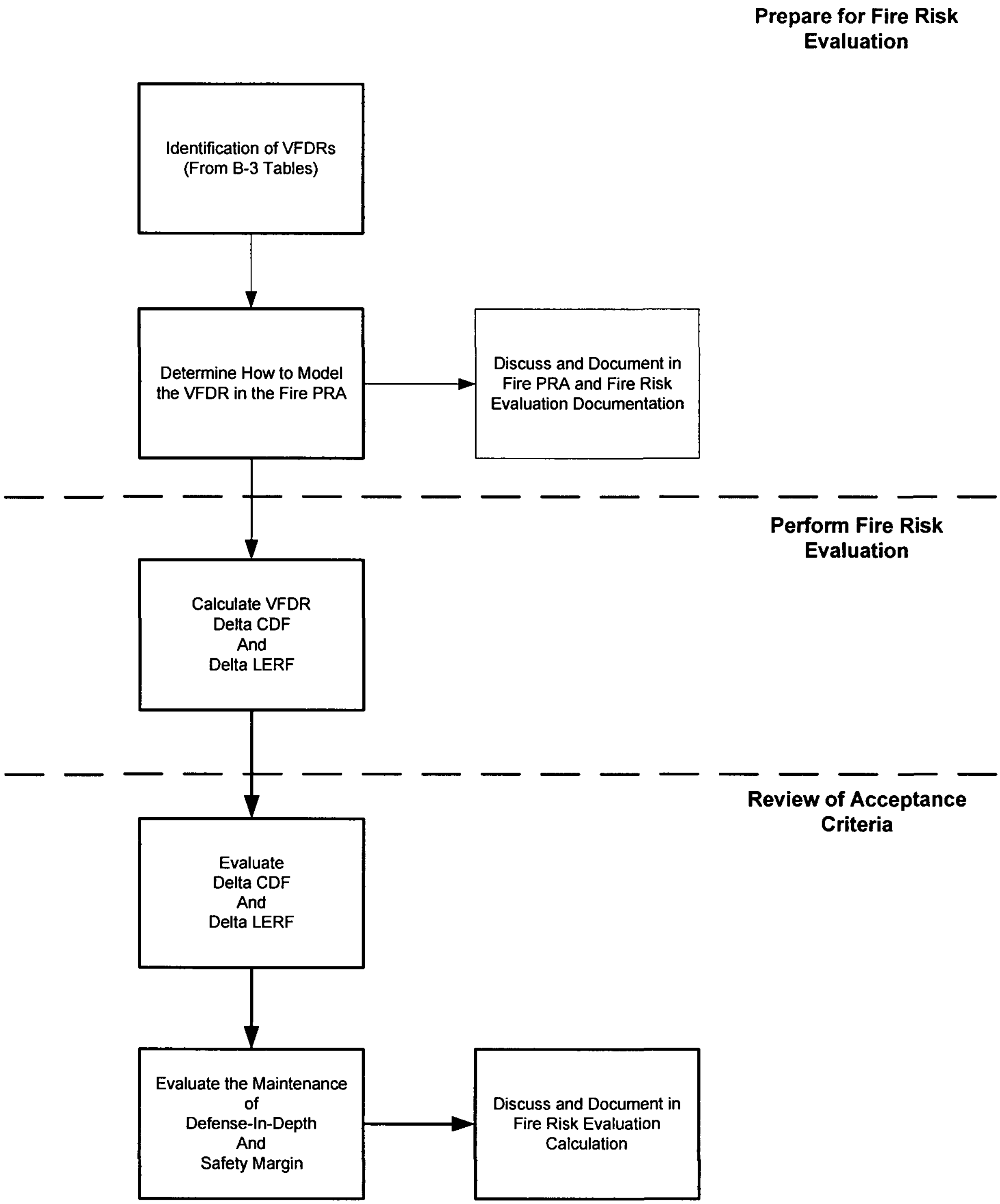

Figure 4-7 – Fire Risk Evaluation Process (NFPA 805 Transition) [Based on FAQ 07-0054 Revision 1]

Results of Evaluation Process

Disposition of VFDRs

The BSEP existing post-fire SSA / NSCA and the NFPA 805 transition project activities have identified a number of variances from the deterministic requirements of NFPA 805 Section 4.2.3. These variances were dispositioned using the fire risk evaluation process.

Each variance dispositioned using a Fire Risk Evaluation was assessed against the Fire Risk Evaluation acceptance criteria of ΔCDF and ΔLERF; and maintenance of defense-in-depth and safety margin criteria from Section 5.3.5 of NEI 04-02 and RG 1.205. The results of these calculations are summarized in Attachment C.

Following completion of transition activities and planned modifications and program changes, the plant will be compliant with 10 CFR 50.48(c).

Risk Change Due to NFPA 805 Transition

In accordance with the guidance in RG 1.205, Section C.2.2.4, Risk Evaluations, risk increases or decreases for each fire area using Fire Risk Evaluations and the overall plant should be provided. Note that the risk increase due to the use of recovery actions was included in the risk change for transition for each fire area.

RG 1.205 Section C.2.2.4.2 states in part

> *"The total increase or decrease in risk associated with the implementation of NFPA 805 for the overall plant should be calculated by summing the risk increases and decreases for each fire area (including any risk increases resulting from previously approved recovery actions). The total risk increase should be consistent with the acceptance guidelines in Regulatory Guide 1.174. Note that the acceptance guidelines of Regulatory Guide 1.174 may require the total CDF, LERF, or both, to evaluate changes where the risk impact exceeds specific guidelines. If the additional risk associated with previously approved recovery actions is greater than the acceptance guidelines in Regulatory Guide 1.174, then the net change in total plant risk incurred by any proposed alternatives to the deterministic criteria in NFPA 805, Chapter 4 (other than the previously approved recovery actions), should be risk neutral or represent a risk decrease."*

The risk increases and decreases are provided in Attachment W.

4.6 Monitoring Program

4.6.1 Overview of NFPA 805 Requirements and NEI 04-02 Guidance on the NFPA 805 Fire Protection System and Feature Monitoring Program

Section 2.6 of NFPA 805 states:

> *"A monitoring program shall be established to ensure that the availability and reliability of the fire protection systems and features are maintained and to assess the performance of the fire protection program in meeting the performance criteria. Monitoring shall ensure that the assumptions in the engineering analysis remain valid."*

As part of the transition review, the adequacy of the inspection and testing program to address fire protection systems and equipment within plant inspection and the compensatory measures programs should be reviewed. In addition, the adequacy of the plant corrective action program in determining the causes of equipment and programmatic failures and minimizing their recurrence should also be reviewed as part of the transition to a risk-informed, performance-based licensing basis.

4.6.2 Overview of Post-Transition NFPA 805 Monitoring Program

This section describes the process that will be utilized to implement the post-transition NFPA 805 monitoring program. The monitoring program will be implemented after the safety evaluation issuance as part of the fire protection program transition to NFPA 805. See item for implementation in Attachment S. The monitoring process is comprised of four phases.

- Phase 1 – Scoping
- Phase 2 – Screening Using Risk Criteria
- Phase 3 – Risk Target Value Determination
- Phase 4 – Monitoring Implementation

Figure 4-8 provides detail on the Phase 1 and 2 processes.

The results of these phases will be documented in the NFPA 805 Monitoring Program scoping document developed during implementation.

Phase 1 – Scoping

In order to meet the NFPA 805 requirements for monitoring, the following categories of SSCs and programmatic elements will be reviewed during the implementation phase for inclusion in the NFPA 805 monitoring program:

- Structures, Systems, and Components required to comply with NFPA 805, specifically:
  - Fire protection systems and features
    - Required by the Nuclear Safety Capability Assessment
    - Modeled in the Fire PRA
    - Required by Chapter 3 of NFPA 805
  - Nuclear Safety Capability Assessment equipment[4]
    - Nuclear safety equipment
    - Fire PRA equipment
    - NPO equipment
  - Structures, systems and components relied upon to meet radioactive release criteria
- Fire Protection Programmatic Elements

[4] For the purposes of the NFPA 805 Monitoring, "NSCA equipment" is intended to include Nuclear Safety Equipment, Fire PRA equipment, and NPO equipment.

Phase 2 – Screening Using Risk Criteria

The equipment from Phase 1 scoping will be screened to determine the appropriate level of NFPA 805 monitoring. As a minimum, the SSCs identified in Phase 1 will be part of an inspection and test program and system/program health reporting. If not in the current program, the SSCs will be added in order to assure that the criteria can be met reliably.

The following screening process will be used to determine those SSCs that may require additional monitoring beyond normal inspection and test program and system/program health reporting and will be documented in the NFPA 805 Monitoring Program scoping document.

1. Fire Protection Systems and Features

Those fire protection systems and features identified in Phase 1 are candidates for additional monitoring in the NFPA 805 program commensurate with risk significance.

Risk significance is determined at the component, programmatic element, and/or functional level on an individual fire area basis. Compartments smaller than fire areas may be used provided the compartments are independent (i.e., share no fire protection SSCs). If compartments smaller than fire areas are used, the basis will be documented in the calculation, BNP-PSA-082.

The Fire PRA is used to establish the risk significance based on the following screening criteria:

Risk Achievement Worth (RAW) of the monitored parameter ≥ 2.0

(AND) either

Core Damage Frequency (CDF) x (RAW) ≥ 1.0E-7 per year

(OR)

Large Early Release Frequency (LERF) x (RAW) ≥ 1.0E-8 per year

CDF, LERF, and $RAW_{(monitored\ parameter)}$ are calculated for each fire area. The 'monitored parameter' will be established at a level commensurate with the amenability of the parameter to risk measurement (e.g., a fire barrier may be more conducive to risk measurement than an individual barrier penetration) and will be documented in the calculation, BNP-PSA-082.

Fire protection systems and features that meet or exceed the criteria identified above are considered High Safety Significant (HSS) and will be included in the NFPA 805 Monitoring Program The HSS fire protection systems and features not already monitored via an existing inspection and test program and/or in the existing system / program health reporting, as described in procedure EGR-NGGC-0010, will be added to the NFPA 805 Monitoring Program and documented in the NFPA 805 Monitoring Program scoping document.

2. Nuclear Safety Capability Assessment Equipment

Required NSCA equipment, except the NPO scope, identified in Phase 1 will be screened for safety significance using the Fire PRA and the Maintenance Rule guidelines differentiating HSS equipment from Low Safety Significant (LSS) equipment. The screening will also ensure that the Maintenance Rule functions are consistent with the required functions of the NSCA equipment.

HSS NSCA equipment not currently monitored in Maintenance Rule will be added into Maintenance Rule. All NSCA equipment that are not HSS are considered LSS and need not be included in the monitoring program.

For non-power operational modes, the qualitative use of fire prevention to manage fire risk during Higher Risk Evolutions does not lend itself to quantitative risk measurement. Therefore, fire risk management effectiveness is monitored programmatically similar to combustible material controls and other fire prevention programs. Additional monitoring beyond inspection and test programs and system/program health reporting is not considered necessary.

3. SSCs Relied upon for Radioactive Release Criteria

The evaluations performed to meet the radioactive release performance criteria are qualitative in nature. The SSCs relied upon to meet the radioactive release performance criteria are not amenable to quantitative risk measurement. Additionally, since 10 CFR Part 20 limits (i.e., which are lower than releases due to core damage and containment breach) for radiological effluents are not being exceeded, equipment relied upon to meet the radioactive release performance criteria is considered inherently low risk. Therefore, additional monitoring beyond inspection and test programs and system/program health reporting is not considered necessary.

4. Fire Protection Programmatic Elements

Monitoring of programmatic elements is required in order to "assess the performance of the fire protection program in meeting the performance criteria". These programs form the bases for many of the analytical assumptions used to evaluate compliance with NFPA 805 requirements. Programmatic aspects include:

- Prompt Detection, including incipient detection fire watch and hot work fire watch
- Transient Combustible Controls Program Violations against FIR-NGGC-0009
- Fire Brigade Effectiveness including Fire Brigade Response Time, Fire Brigade Fire Drill, and Fire Brigade Fire Drill Objectives

Monitoring of programmatic elements is more qualitative in nature since the programs do not lend themselves to the numerical methods of reliability and availability. Therefore, monitoring is conducted using the existing program health programs. Fire protection health reports, self-assessments, regulator and insurance company reports provide inputs to the monitoring program.

Phase 3 – Risk Target Value Determination

Failure criteria is established by an expert panel based on the required fire protection and nuclear safety capability SSCs and programmatic elements assumed level of performance in the supporting analyses established in Phase 2. Action levels are established for the SSCs at the component level, program level, or functionally through the use of the pseudo system or 'performance monitoring group' concept. The actual action level is determined based on the number of component, program or functional failures within a sufficiently bounding time period (i.e., ~2-3 operating cycles).

Since the HSS NSCA equipment have been identified using the Maintenance Rule guidelines, the associated equipment specific performance criteria will be established as in the Maintenance Rule, provided the criteria are consistent with Fire PRA assumptions.

When establishing the action level threshold for reliability and availability, the action level will be no lower than the fire PRA assumptions. Adverse trends and unacceptable levels of availability, reliability, and performance will be reviewed against established action levels.

Documentation of the monitoring program failure criteria and action level targets will be contained in a documented evaluation. It is anticipated that the availability and reliability criterion for High Safety Significant Performance Monitoring Groups will use the guidance included in several industry documents tempered by site-specific operating experience, Fire PRA assumptions, and equipment types (and vendor data or valid design input when available). Industry documents such as the EPRI Fire Protection Equipment Surveillance Optimization and Maintenance Guide TR-1006756, Final Report July 2003, NFPA codes, and/or the NRC Fire Protection Significance Determination Process in addition to site specific operating experience data may be used. The monitoring program failure criteria and action level targets will be documented in the NFPA 805 Monitoring Program scoping document.

Note that fire protection systems and features, NSCA equipment, SSCs required to meet the radioactive release criteria, and fire protection program elements that do not meet the screening criteria in Phase 2 will be included in the existing inspection and test programs and the system and program health programs. Reliability and availability criteria will not be assigned.

Phase 4 – Monitoring Implementation

Phase 4 is the implementation of the monitoring program, once the monitoring scope and criteria are established. Monitoring consists of periodically gathering, trending, and evaluating information pertinent to the performance, and/or availability of the equipment and comparing the results with the established goals and performance criteria to verify that the goals are being met. Results of monitoring activities will be analyzed in a timely manner to assure that appropriate action is taken. The corrective action process will be used to address performance of fire protection and nuclear safety SSCs that do not meet performance criteria.

For fire protection systems and features and NSCA HSS equipment that are monitored, unacceptable levels of availability, reliability, and performance will be reviewed against the established action levels. If an action level is triggered, corrective action in accordance with procedure, CAP-NGGC-0200 will be initiated to identify the negative trend. A corrective action plan will then be developed to ensure the performance returns to the established level.

When applicable, a sensitivity study can be performed to determine the margin below the action level that still provides acceptable fire PRA results to help prioritize corrective actions if the action level is reached.

A periodic assessment will be performed (i.e., at a frequency of approximately every two to three operating cycles), taking into account, where practical, industry wide operating experience. Issues that will be addressed include:

- Review systems with performance criteria. Do performance criteria still effectively monitor the functions of the system? Do the criteria still monitor the effectiveness of the fire protection and NSCA systems?
- Have the supporting analyses been revised such that the performance criteria are no longer applicable or new fire protection and NSCA SSCs, programmatic elements and/ or functions need to be in scope?
- Based on the performance during the assessment period, are there any trends in system performance that should be addressed that are not being addressed?

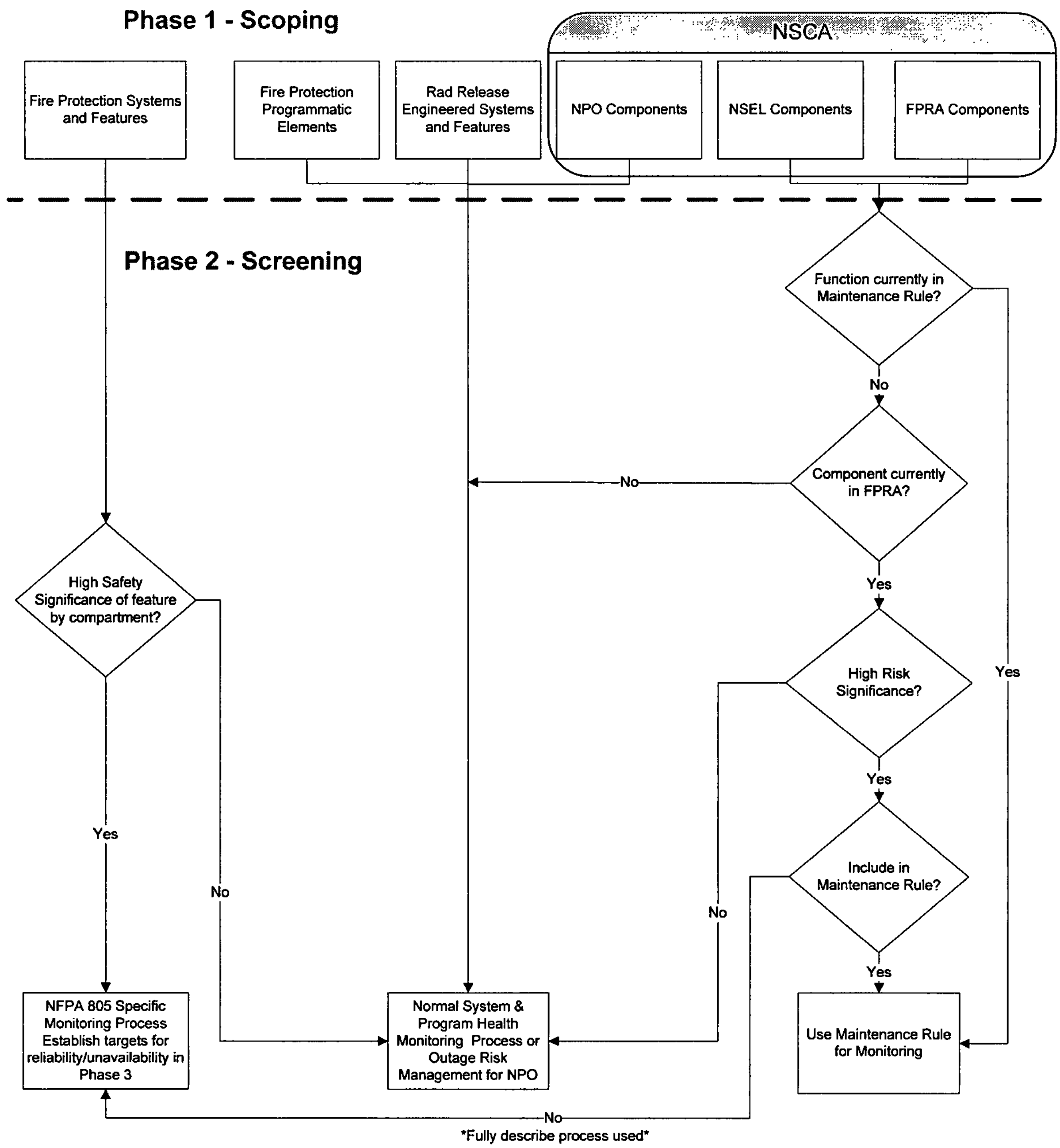

Figure 4-8 – NFPA 805 Monitoring – Scoping and Screening

4.7 Program Documentation, Configuration Control, and Quality Assurance

4.7.1 Compliance with Documentation Requirements in Section 2.7.1 of NFPA 805

In accordance with the requirements and guidance in NFPA 805 Section 2.7.1, and NEI 04-02, BSEP has documented analyses to support compliance with

10 CFR 50.48(c). The analyses are being performed in accordance with CP&L's processes for ensuring assumptions are clearly defined, that results are easily understood, that results are clearly and consistently described, and that sufficient detail is provided to allow future review of the entire analyses.

Analyses, as defined by NFPA 805 Section 2.4, performed to demonstrate compliance with 10 CFR 50.48(c) will be maintained for the life of the plant and organized to facilitate review for accuracy and adequacy. Note these analyses do not include items such as periodic tests, hot work permits, fire impairments, etc.

The Fire Protection Design Basis Document described in Section 2.7.1.2 of NFPA 805 and necessary supporting documentation described in Section 2.7.1.3 of NFPA 805 have been created as part of transition to 10 CFR 50.48(c) to ensure program implementation following receipt of the safety evaluation. Figure 4-9 shows the Planned Post-Transition Documents.

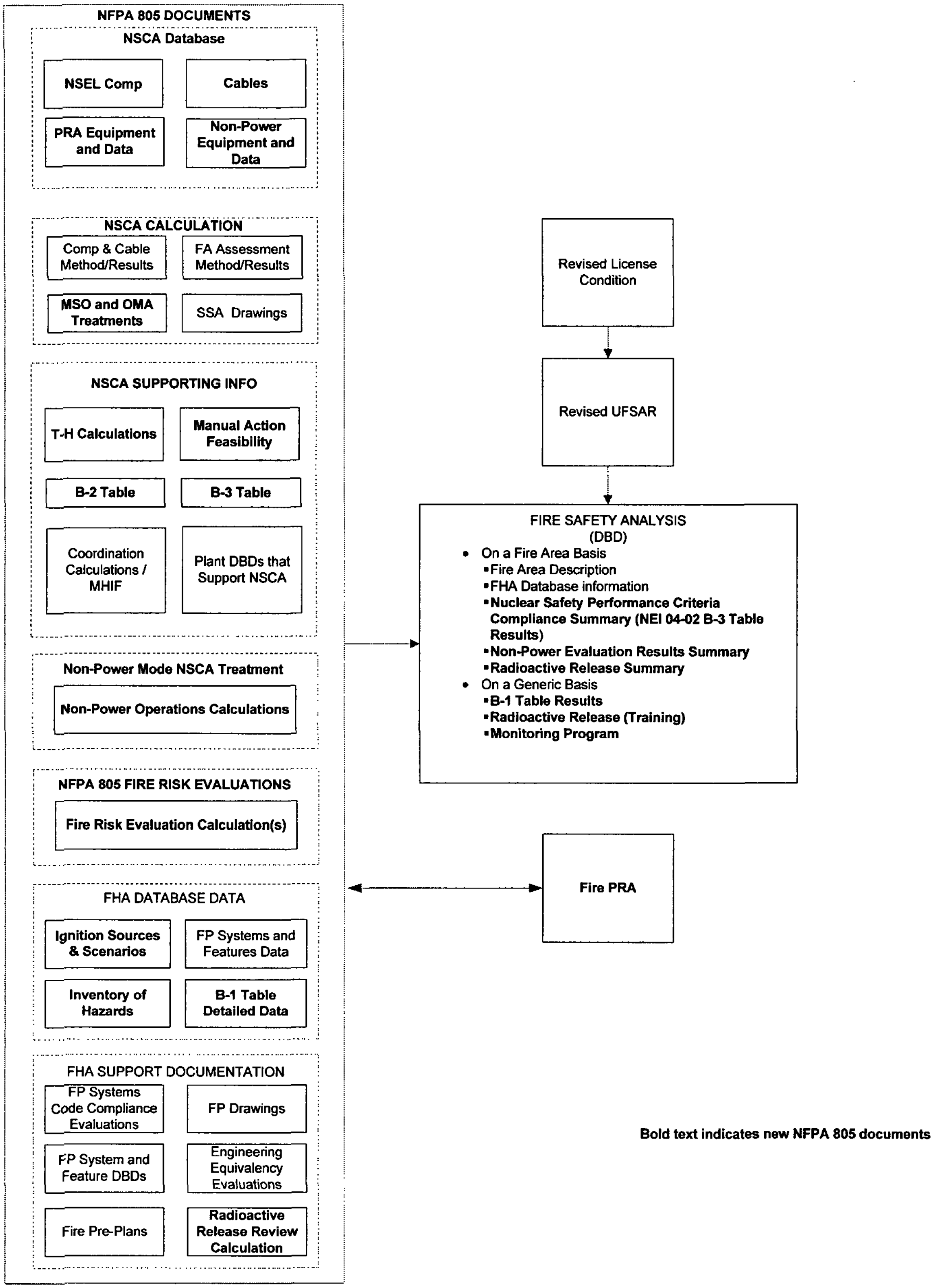

Figure 4-9 – NFPA 805 Planned Post-Transition Documents and Relationships

4.7.2 Compliance with Configuration Control Requirements in Section 2.7.2 and 2.2.9 of NFPA 805

Program documentation established, revised, or utilized in support of compliance with 10 CFR 50.48(c) is subject to CP&L configuration control processes that meet the requirements of Section 2.7.2 of NFPA 805. This includes the appropriate procedures and configuration control processes for ensuring that changes impacting the fire protection program are reviewed appropriately. The RI-PB post transition change process methodology is based upon the requirements of NFPA 805, and industry guidance in NEI 04-02, and RG 1.205. These requirements are summarized in Table 4-2.

Table 4-2 Change Evaluation Guidance Summary Table

Document	Section(s)	Topic
NFPA 805	2.2(h), 2.2.9, 2.4.4, A.2.2(h), A.2.4.4, D.5	Change Evaluation
NEI 04-02	5.3, Appendix B, Appendix I, Appendix J	Change Evaluation, Change Evaluation Forms (Appendix I)
RG 1.205	C.2.2.4, C.3.1, C.3.2, C.4.3	Risk Evaluation, Standard License Condition, Change Evaluation Process, Fire PRA

The Plant Change Evaluation Process consists of the following 4 steps and is depicted in Figure 4-10:

- Defining the Change
- Performing the Preliminary Risk Screening.
- Performing the Risk Evaluation
- Evaluating the Acceptance Criteria

Change Definition

The Change Evaluation process begins by defining the change or altered condition to be examined and the baseline configuration as defined by the Design Basis and Licensing Basis (NFPA 805 Licensing Basis post-transition).

1. The baseline is defined as that plant condition or configuration that is consistent with the Design Basis and Licensing Basis (NFPA 805 Licensing Basis post-transition).
2. The changed or altered condition or configuration that is not consistent with the Design Basis and Licensing Basis is defined as the proposed alternative.

Preliminary Risk Review

Once the definition of the change is established, a screening is then performed to identify and resolve minor changes to the fire protection program. This screening is consistent with fire protection regulatory review processes in place at nuclear plants under traditional licensing bases. This screening process is modeled after the NEI 02-03 process. This process will address most administrative changes (e.g., changes to the combustible control program, organizational changes).

The characteristics of an acceptable screening process that meets the "assessment of the acceptability of risk" requirement of Section 2.4.4 of NFPA 805 are:

- The quality of the screen is sufficient to ensure that potentially greater than minimal risk increases receive detailed risk assessments appropriate to the level of risk.
- The screening process must be documented and be available for inspection by the NRC.
- The screening process does not pose undue evaluation or maintenance burden.

If any of the above is not met, proceed to the Risk Evaluation step.

Risk Evaluation

The screening is followed by engineering evaluations. The results of these evaluations are then compared to the acceptance criteria. Changes that satisfy the acceptance criteria of NFPA 805 Section 2.4.4 and the license condition can be implemented within the framework provided by NFPA 805. Changes that do not satisfy the acceptance criteria cannot be implemented within this framework. The acceptance criteria require that the resultant change in CDF and LERF be consistent with the license condition. The acceptance criteria also include consideration of defense-in-depth and safety margin, which would typically be qualitative in nature.

The risk evaluation involves the application of risk assessment techniques to obtain a measure of the changes in risk associated with the proposed change. In certain circumstances, an initial evaluation in the development of the risk assessment could be a simplified analysis using bounding assumptions provided the use of such assumptions does not unnecessarily challenge the acceptance criteria discussed below.

Acceptability Determination

The Change Evaluations are assessed for acceptability using the ΔCDF (i.e., change in core damage frequency) and ΔLERF (i.e., change in large early release frequency) criteria from the license condition. The proposed changes are also assessed to ensure they are consistent with the defense-in-depth philosophy and that sufficient safety margins were maintained.

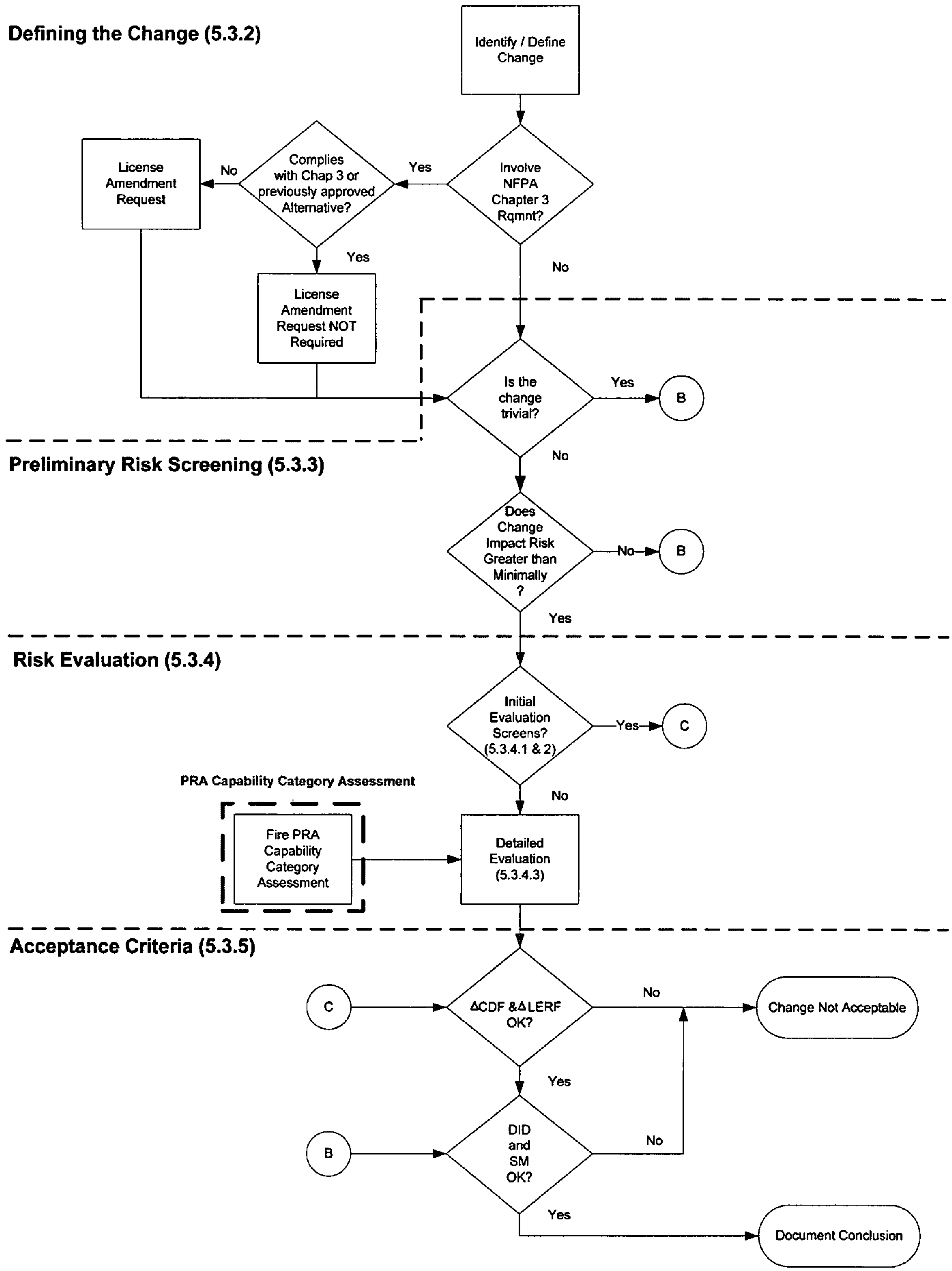

Figure 4-10 Plant Change Evaluation [NEI 04-02 Figure 5-1]
Note references in Figure refer to NEI 04-02 Sections

The BSEP Fire Protection Program configuration is defined by the program documentation. The existing configuration control processes for modifications, calculations and analyses, and Fire Protection Program License Basis Reviews will be utilized to maintain configuration control of the Fire Protection program documents. The configuration control procedures which govern the various BSEP documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements (Implementation Item in Attachment S).

Several NFPA 805 document types, such as NSCA Supporting Information, Non-Power Mode NSCA Treatment, generally require new control procedures and processes to be developed since they are new documents and databases created as a result of the transition to NFPA 805. The new procedures will be modeled after the existing processes for similar types of documents and databases. System level design basis documents will be revised to reflect the NFPA 805 role that the system components now play.

The process for capturing the impact of proposed changes to the plant on the Fire Protection Program will continue to be a multiple step review. The first step of the review is an initial screening for process users to determine if there is a potential to impact the Fire Protection program as defined under NFPA 805 through a series of screening questions/checklists contained in one or more procedures depending upon the configuration control process being used. Reviews that identify potential Fire Protection program impacts will be sent to qualified individuals (Fire Protection, Safe Shutdown/NSCA, Fire PRA) to ascertain the program impacts, if any. If Fire Protection program impacts are determined to exist as a result of the proposed change, the issue would be resolved by one of the following:

- Deterministic Approach: Comply with NFPA 805, Chapter 3 and 4.2.3 requirements
- Performance-Based Approach: Utilize the NFPA 805 change process developed in accordance with NEI 04-02, RG 1.205, and the NFPA 805 fire protection license condition to assess the acceptability of the proposed change. This process would be used to determine if the proposed change could be implemented "as-is" or whether prior NRC approval of the proposed change is required.

This process follows the requirements in NFPA 805 and the guidance outlined in RG 1.174, which requires the use of qualified individuals, procedures that require calculations be subject to independent review and verification, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when errors are discovered. The plant documents that ensure these requirements are met are:

CAP-NGGC-0200 – Condition Identification and Screening Process
EGR-NGGC-0005 – Engineering Change
ESG0101N – Safe Shutdown Engineer (Post-NFPA 805 Transition)
ESG0102N – Fire Protection Plant Change Impact Review
ESG0103N – Circuit Analysis (Post-NFPA 805 Transition)

ESG0104N – Fire Protection Engineer (Post-NFPA 805 Transition)
ESG0105N – Basic Fire Modeling

4.7.3 Compliance with Quality Requirements in Section 2.7.3 of NFPA 805

Fire Protection Program Quality

CP&L will maintain the existing fire protection quality assurance program.

During the transition to 10 CFR 50.48(c), BSEP performed work in accordance with the quality requirements of Section 2.7.3 of NFPA 805.

Fire PRA Quality

Configuration control of the Fire PRA model will be maintained by integrating the Fire PRA model into the existing processes used to ensure configuration control of the internal events PRA model. This process complies with Section 1-5 of the ASME PRA Standard and ensures that CP&L maintains an as-built, as-operated PRA model of the plant. The process has been peer reviewed. Quality assurance of the Fire PRA is assured via the same processes applied to the internal events model.

This process follows the guidance outlined in RG 1.174, which requires the use of qualified individuals, procedures that require calculations be subject to independent review and verification, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when errors are discovered. Although the entire scope of the formal 10 CFR 50, Appendix B program is not applied to the PRA models or processes in general, often parts of the program are applied as a convenient method of complying with the requirements of RG 1.174. For instance, the procedure which addresses independent review of calculations for 10 CFR 50, Appendix B, is applied to the PRA model calculations, as well.

With respect to Quality Assurance Program requirements for independent reviews of calculations and evaluations, those existing requirements for Fire Protection Program documents will remain unchanged. CP&L specifically requires that the calculations and evaluations in support of the NFPA 805 LAR, exclusive of the Fire PRA, be performed within the scope of the QA program which requires independent review as defined by plant procedures. As recommended by NUREG/CR-6850, the sources of uncertainty in the Fire PRA were identified and specific parameters were analyzed for sensitivity in support of the NFPA 805 Fire Risk Evaluation process.

Specifically with regard to uncertainty, an uncertainty and sensitivity matrix was developed and included with BNP-PSA-080. In addition, sensitivity to uncertainty associated with specific Fire PRA parameters was quantitatively addressed in BNP-PSA-095.

While the removal of conservatism inherent in the Fire PRA is a long-term goal, the Fire PRA results were deemed sufficient for evaluating the risk associated with this application. While CP&L continues to strive toward a more "realistic" estimate of fire risk, use of mean values continues to be the best estimate of fire risk. During the Fire Risk Evaluation process, the uncertainty and sensitivity associated with specific Fire PRA parameters were considerations in the evaluation of the change in risk relative to the applicable acceptance thresholds.

Specific Requirements of NFPA 805 Section 2.7.3

NFPA 805 Section 2.7.3.1 – Review

Analyses, calculations, and evaluations performed in support of compliance with 10 CFR 50.48(c) are performed in accordance with procedures that require independent review.

Reference plant procedures:

EGR-NGGC-0003 – Design Review Requirements

EGR-NGGC-0005 – Engineering Change

EGR-NGGC-0017 – Preparation and Control of Design Analyses and Calculations

NFPA 805 Section 2.7.3.2 – Verification and Validation

Calculational models and numerical methods used in support of compliance with 10 CFR 50.48(c) were verified and validated as required by Section 2.7.3.2 of NFPA 805.

NFPA 805 Section 2.7.3.3 – Limitations of Use

Engineering methods and numerical models used in support of compliance with 10 CFR 50.48(c) were applied appropriately as required by Section 2.7.3.3 of NFPA 805.

NFPA 805 Section 2.7.3.4 – Qualification of Users

Cognizant personnel who use and apply engineering analysis and numerical methods in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by Section 2.7.3.4 of NFPA 805.

During the transition to 10 CFR 50.48(c), work was performed in accordance with the quality requirements of Section 2.7.3 of NFPA 805. Personnel who used and applied engineering analysis and numerical methods (e.g., fire modeling) in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by NFPA 805, Section 2.7.3.4.

Post-transition, for personnel performing fire modeling or Fire PRA development and evaluation, CP&L has developed and maintains qualification requirements for individuals assigned various tasks. Position-Specific Guides have been developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per the requirements of NFPA 805, Section 2.7.3.4 to perform assigned work. The following Training Guides have been developed and implemented.

ESG0089N - Fire Probabilistic Safety Assessment Engineer (Quantification),
ESG0093N - Fire Probabilistic Safety Assessment Engineer (Initial Development), and
ESG0094N - Fire Probabilistic Safety Assessment Engineer (Data Development), and
ESG0105N – Basic Fire Modeling

NFPA 805 Section 2.7.3.5 – Uncertainty Analysis

Uncertainty analyses were performed as required by 2.7.3.5 of NFPA 805 and the results were considered in the context of the application. This is of particular interest in fire modeling and Fire PRA development. Note: 10 CFR 50.48(c)(2)(iv) states that NFPA 805, Section 2.7.3.5 is not required for the deterministic approach because conservatism is included in the deterministic criteria.

4.8 Summary of Results

4.8.1 Results of the Fire Area Review

A summary of the NFPA 805 compliance basis and the required fire protection systems and features is provided in Attachment C. The table provides the following information from the NEI 04-02 Table B-3:

- Fire Area / Fire Zone: Fire Area/Zone Identifier.
- Description: Fire Area/Zone Description.
- NFPA 805 Regulatory Basis: Post-transition NFPA 805 Chapter 4 compliance basis
- Required Fire Protection System / Feature: Detection / suppression required in the Fire Area based on NFPA 805 Chapter 4 compliance. Other Required Features may include Electrical Raceway Fire Barrier Systems, fire barriers, etc. The documentation of required fire protection systems and features does not include the documentation of the fire area boundaries. Fire area boundaries are required and documentation of the fire area boundaries has been performed as part of reviews of engineering evaluations, licensing actions, or as part of the reviews of the NEI 04-02 Table B-1 process. The basis for the requirement of the fire protection system / feature is designated as follows:
  - S – Separation Criteria: Systems/Features required for Chapter 4 Separation Criteria in Section 4.2.3
  - E – EEEE/LA Criteria: Systems/Features required for acceptability of Existing Engineering Equivalency Evaluations / NRC approved Licensing Action (i.e., Exemptions/Deviations/Safety Evaluations) (Section 2.2.7)
  - R – Risk Criteria: Systems/Features required to meet the Risk Criteria for the Performance-Based Approach (Section 4.2.4)
  - D – Defense-in-depth Criteria: Systems/Features required to maintain adequate balance of Defense-in-Depth for a Performance-Based Approach (Section 4.2.4)

An evaluation of DID was performed for all fire areas as detailed in project procedure FPIP-129, NFPA 805 Fire Safety Analysis. This evaluation was performed for all areas, regardless of whether NFPA 805 compliance was demonstrated using a performance based approach or a deterministic approach. Although a discussion of DID features is not strictly required for areas that are deterministically compliant, the decision to include the evaluation for such areas was based on two factors. First, it was seen as a way of enhancing the overall approach to providing the plant's desired level of fire protection to that area. Second, if future changes to deterministic areas dictate that a performance based approach is desired, then including these features as credited DID features now will facilitate that transition. The regulatory basis for each fire area is provided in

Attachment C, but the presence of deterministic features in the DID discussion does not alter any conclusions regarding the post transition licensing basis.

Attachment W contains the results of the Fire Risk Evaluations, additional risk of recovery actions, and the change in risk on a fire area basis.

4.8.2 Plant Modifications and Items to be Completed During the Implementation Phase

Planned modifications, studies, and evaluations to comply with NFPA 805 are described in Attachment S.

In Attachment S, two tables are listed. Table S-1 identifies Plant Modifications required to be completed. Table S-2 identifies training, programs, personnel equipment, and document changes and upgrades required to be completed.

The Fire PRA model will represent the as-built, as-operated and maintained plant following completion of the risk related modifications identified in Attachment S. In the event the PRA model requires revision following completion of the modifications and implementation items listed in Attachment S, the changes will be controlled through normal BSEP processes. These changes are not expected to be significant. The Main Control Room ceiling modification is the only outstanding change with respect to its inclusion in the Fire PRA model.

4.8.3 Supplemental Information –Other Licensee Specific Issues

The development of a FPRA requires that assumptions and methods be expanded and updated to provide more realistic treatment of the data and the phenomena involved. The updates and expansion of methods introduce differences in plant specific results depending on which alternatives are used. This section captures the sensitivities and insights based on these alternatives. These alternatives may be based on new analysis methods, new data, or deviations from guidance in NUREG/CR-6850 which would require sensitivity analyses to be performed for the license application.

4.8.3.1 Unreviewed Analysis Methods

The peer review of the BSEP FPRA identified one method that had not been reviewed by the methods panel concerning the use of a split fraction for closed cabinet fires that result in damage outside of the cabinet. This method was reviewed by the NRC for the Harris plant NFPA 805 pilot effort as documented in Section 3.4.7 and Table 3.4-6 of the Safety Evaluation for the Harris Plant license amendment (ML101130535).

There is variation in the methods in treating how MCCs can be treated as "closed" cabinets. If a cabinet were always "closed" there would be no fire impact on external targets. However, there is always the potential for the cabinet to already be open or an arc fault to have enough energy to open the cabinet. For the BSEP FPRA, it was assumed that one out of ten MCC fires may result in an "open" cabinet configuration. This is not applied to the HRR as a severity factor, but as a split fraction on the likelihood that the cabinet remains "closed."

The guidance for characterizing closed cabinets at the Brunswick plant was the same as that used for the Harris Nuclear Plant pilot effort. The use of split fractions as described above was deemed acceptable for use by the NRC at HNP.

A sensitivity analysis was performed on this method for the BSEP FPRA. The sensitivity analysis essentially removed the split fraction, effectively treating the closed MCCs as always open. The results of the sensitivity study for the "closed" cabinet method are provided below.

Table 4-4 - Closed MCCs Sensitivity Delta CDF and Delta LERF Results

	Unit 1		Unit 2	
	ΔCDF [/yr]	ΔLERF [/yr]	ΔCDF [/yr]	ΔLERF [/yr]
VFDRs	1.2E-06	1.3E-08	2.7E-06	5.2E-09
Recovery Actions	9.1E-07	9.1E-08	9.1E-07	9.1E-08
Total	2.1E-06	1.0E-07	3.6E-06	9.6E-08

A manual summation may differ from the Total due to rounding in the last digit.

Table 4-5 - Closed MCCs Sensitivity Total CDF and Total LERF Results

	Unit 1		Unit 2	
	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]
Internal Events plus External Flooding and High Winds	1.4E-05	6.2E-07	1.4E-05	6.2E-07
Fire[1]	2.1E-05	4.3E-06	2.0E-05	4.0E-06
Fire – Recovery Actions[2]	1.0E-06	1.0E-07	1.0E-06	1.0E-07
Total	3.6E-05	5.0E-06	3.5E-05	4.8E-06

[1] Fire results do not credit control room abandonment for loss of control sequences.

[2] Values are for recovery actions associated with control room abandonment due to environmental reasons.

A manual summation may differ from the Total due to rounding in the last digit.

4.8.3.2 Concerns with NUREG/CR-6850 CPT Credit

Based on preliminary results for fire circuit testing, the credit allowed in Tables 10-1 and 10-3 of NUREG/CR-6850 for Control Power Transformers (CPT) in AC circuits was questioned by NRR. This is based on an RAI letter to Duane Arnold (ML12031A112). The sensitivity analysis was performed by removing the approximate factor of two reduction in failure mode probability estimates between cables with CPT and those without. The results of the sensitivity study analysis for the CPT credit are provided below.

Table 4-6 - CPT Sensitivity Delta CDF and Delta LERF Results

	Unit 1		Unit 2	
	ΔCDF [/yr]	ΔLERF [/yr]	ΔCDF [/yr]	ΔLERF [/yr]
VFDRs	1.2E-06	1.4E-08	2.7E-06	5.2E-09
Recovery Actions	9.1E-07	9.1E-08	9.1E-07	9.1E-08
Total	2.1E-06	1.0E-07	3.6E-06	9.6E-08

A manual summation may differ from the Total due to rounding in the last digit.

Table 4-7 - CPT Sensitivity Total CDF and Total LERF

	Unit 1		Unit 2	
	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]
Internal Events plus External Flooding and High Winds	1.4E-05	6.2E-07	1.4E-05	6.2E-07
Fire[1]	2.1E-05	4.3E-06	2.0E-05	4.0E-06
Fire – Recovery Actions[2]	1.0E-06	1.0E-07	1.0E-06	1.0E-07
Total	3.6E-05	5.0E-06	3.5E-05	4.8E-06

[1] Fire results do not credit control room abandonment for loss of control sequences.

[2] Values are for recovery actions associated with control room abandonment due to environmental reasons.

A manual summation may differ from the Total due to rounding in the last digit.

4.8.3.3 Sensitivity Analysis Required by FAQ 08-0048

In order to use the updated fire bin ignition frequencies provided in Supplement 1 to NUREG/CR-6850, a sensitivity analysis must be performed comparing the impact of those bins characterized by an alpha from the EPRI TR-1016735 analysis that is less than or equal to 1. While the new point estimates for the bin ignition frequencies better represent the data, uncertainties are large and a sensitivity analysis using the older conservative frequencies is required to assess the potential impact of using the new frequencies.

Table 4-8 - Ignition Frequency Sensitivity Delta CDF and Delta LERF Results

	Unit 1		Unit 2	
	ΔCDF [/yr]	ΔLERF [/yr]	ΔCDF [/yr]	ΔLERF [/yr]
VFDRs	2.0E-06	2.2E-08	3.6E-06	6.9E-09
Recovery Actions	2.7E-06	2.7E-07	2.7E-06	2.7E-07
Total	4.6E-06	2.9E-07	6.3E-06	2.7E-07

A manual summation may differ from the Total due to rounding in the last digit.

Table 4-9 - Ignition Frequency Sensitivity Total CDF and Total LERF

	Unit 1		Unit 2	
	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]
Internal Events plus External Flooding and High Winds	1.4E-05	6.2E-07	1.4E-05	6.2E-07
Fire[1]	3.4E-05	8.6E-06	3.6E-05	8.3E-06
Fire – Recovery Actions[2]	3.1E-06	3.1E-07	3.1E-06	3.1E-07
Total	5.1E-05	9.5E-06	5.3E-05	9.2E-06

[1] Fire results do not credit control room abandonment for loss of control sequences.

[2] Values are for recovery actions associated with control room abandonment due to environmental reasons.

A manual summation may differ from the Total due to rounding in the last digit.

4.8.3.4 Main Control Room Abandonment

The control room abandonment has a detailed human error analysis.

Control room abandonment was not credited for loss of control scenarios.

The Brunswick control room has two main areas that contribute to control room abandonment: the area where in the control room staff generally manipulates controls and, the area that is outside that region. The region that the control room staff generally occupies while manipulating the controls has a much lower ceiling height and smaller footprint than the remaining area and, as such, has a much shorter time to suppress a fire prior to reaching control room habitability concerns that require abandonment. A shorter time to suppress the fire increases the frequency of control room abandonment which is directly proportional to CDF and LERF contributions for control room abandonment. The control room abandonment risk is conservative since the fire frequency contribution from the cabinets that comprise the boundary for the control manipulation area is all applied to the small area when there is significant probability that the fire would vent out of the back of the panels to the larger area with higher ceilings. Since methods to determine a split fraction of fires that vent to the rear of the panel were not peer reviewed, all of the frequency was conservatively applied to the smaller region with the low ceiling resulting in conservative times to conditions requiring control room abandonment.

4.8.3.5 Reduction in Transient Source Heat Release Rate

Following transition to NFPA 805, BSEP will adopt a more restrictive transient control program that will nominally limit the transient fire HRR to the 143 kW range instead of the 317 kW range. The transient control program is the fleet program and is already in use at HNP. The 143 kW range was used for the transient fire locations in all areas except for the turbine building, which uses a 317 kW HRR.

4.8.3.6 Incipient Detection in Main Control Boards

The FPRA credits the use of air-aspirated incipient detection, also known as Very Early Warning Fire Detection Systems (VEWFDS) in NFPA 76, in the Main Control Boards (MCBs) because that modification is expected to be completed prior to the transition to NFPA 805. To support the use of incipient detection, a walkdown was performed for a representative sample of BSEP MCBs and determined the fraction of fast-acting components to be very small (less than 0.5%) of the total component count.

A sensitivity analysis was performed using the currently installed in-panel ion smoke detection rather than the incipient detection. The NUREG/CR-6850 Appendix L method was used to determine the frequency of self-fires that cause fire damage only within the MCBs, then the ignition frequency of NUREG/CR-6850 Supplement 1 was modified by the "normal" non-suppression probability for ion smoke detectors and manual detection/suppression for fires that also cause damage in the zone-of-influence outside the MCBs. For the zone-of-influence fires, the human reliability analysis for LERF was further refined to account for operator actions to secure AC and DC power to primary containment isolation valves during MCR abandonment not related to habitability issues.

Table 4-10 – MCB Incipient Detection Sensitivity Delta CDF and Delta LERF Results

	Unit 1		Unit 2	
	ΔCDF [/yr]	ΔLERF [/yr]	ΔCDF [/yr]	ΔLERF [/yr]
VFDRs	1.2E-06	1.3E-08	2.7E-06	5.2E-09
Recovery Actions [1]	9.1E-07	9.1E-08	9.1E-07	9.1E-08
Total	2.1E-06	1.0E-07	3.6E-06	9.6E-08

[1] Values are for recovery actions associated with control room abandonment due to environmental reasons and address those actions away from the remote shutdown panel.

A manual summation may differ from the Total due to rounding in the last digit.

Table 4-11 – MCB Incipient Detection Sensitivity Total CDF and Total LERF Results

	Unit 1		Unit 2	
	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]
Internal Events plus External Flooding and High Winds	1.4E-05	6.2E-07	1.4E-05	6.2E-07
Fire[1]	4.8E-05	5.6E-06	5.1E-05	5.5E-06
Fire – Recovery Actions[2]	1.0E-06	1.0E-07	1.0E-06	1.0E-07
Total	6.3E-05	6.3E-06	6.6E-05	6.2E-06

[1] Fire results credit operator actions to secure power during control room abandonment for loss of control sequences.

[2] Values are for recovery actions associated with control room abandonment due to environmental reasons.

A manual summation may differ from the Total due to rounding in the last digit.

4.8.3.7 Pump Heat Release Rate for Motor Pumps

The use of the 98th percentile HRR of 69kW for motor fires from NUREG/CR-6850 as opposed to the electrical pump fire HRR of 211 kW was questioned by NRR. The sensitivity study changed the HRR to 211kW for motor fires, which resulted in additional targets. The results of the sensitivity study are provided below.

Table 4-12 - Pump HRR for Motor Pumps Sensitivity Delta CDF and Delta LERF Results

	Unit 1		Unit 2	
	ΔCDF [/yr]	ΔLERF [/yr]	ΔCDF [/yr]	ΔLERF [/yr]
VFDRs	1.2E-06	1.3E-08	2.7E-06	5.2E-09
Recovery Actions	9.1E-07	9.1E-08	9.1E-07	9.1E-08
Total	2.1E-06	1.0E-07	3.6E-06	9.6E-08

A manual summation may differ from the Total due to rounding in the last digit.

Table 4-13 - Pump HRR for Motor Pumps Sensitivity Total CDF and Total LERF Results

	Unit 1		Unit 2	
	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]
Internal Events plus External Flooding and High Winds	1.4E-05	6.2E-07	1.4E-05	6.2E-07
Fire[1]	2.0E-05	4.3E-06	2.0E-05	4.0E-06
Fire – Recovery Actions[2]	1.0E-06	1.0E-07	1.0E-06	1.0E-07
Total	3.5E-05	5.0E-06	3.5E-05	4.8E-06

[1] Fire results credit operator actions to secure power during control room abandonment for loss of control sequences.
[2] Values are for recovery actions associated with control room abandonment due to environmental reasons.
A manual summation may differ from the Total due to rounding in the last digit.

4.8.3.8 Sensitive Electronics Sensitivity Analysis

A quantitative evaluation has been conducted for determining if the current treatment of fire damage to sensitive electronics in the BSEP FPRA has a significant effect on the quantified fire risk values. The evaluation consisted of two main activities: 1) identifying sensitive electronic components credited in the FPRA, and 2) adding the fire compartments of each identified sources full transient frequency to the IGF for each source containing sensitive electronic equipment. The results of the sensitivity study are provided below.

Table 4-14 – Sensitive Electronics Sensitivity Delta CDF and Delta LERF Results

	Unit 1		Unit 2	
	ΔCDF [/yr]	ΔLERF [/yr]	ΔCDF [/yr]	ΔLERF [/yr]
VFDRs	1.2E-06	1.3E-08	2.7E-06	5.2E-09
Recovery Actions	9.1E-07	9.1E-08	9.1E-07	9.1E-08
Total	2.1E-06	1.0E-07	3.6E-06	9.6E-08

A manual summation may differ from the Total due to rounding in the last digit.

Table 4-15 - Sensitive Electronics Sensitivity Total CDF and Total LERF Results

	Unit 1		Unit 2	
	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]
Internal Events plus External Flooding and High Winds	1.4E-05	6.2E-07	1.4E-05	6.2E-07
Fire[1]	2.0E-05	4.3E-06	2.0E-05	4.0E-06
Fire – Recovery Actions[2]	1.0E-06	1.0E-07	1.0E-06	1.0E-07
Total	3.5E-05	5.0E-06	3.5E-05	4.8E-06

[1] Fire results credit operator actions to secure power during control room abandonment for loss of control sequences.
[2] Values are for recovery actions associated with control room abandonment due to environmental reasons.
A manual summation may differ from the Total due to rounding in the last digit.

4.8.3.9 Smoke Damage Sensitivity Analysis

A qualitative evaluation has been conducted for determining if the current treatment of smoke damage in the BSEP FPRA has a significant effect on the quantified fire risk values (i.e. CDF and LERF). The results of the evaluation suggest that the current treatment of smoke damage in the BSEP FPRA is consistent with the guidance in Appendix T of NUREG/CR-6850. The quantification approach for ignition sources and targets inside and outside of the control room captures failure of the credited function in the FPRA that could be generated by smoke damage (i.e., function failures within the panels connected to the panel of fire origin).

4.8.3.10 Main Control Room Abandonment Bundle Treatment

A sensitivity analysis was done to treat all Main Control Board (MCB) panels as multi-bundle fires, rather than single bundle fires, in the MCR Abandonment study. The risk of abandonment with the single bundle treatment was compared to risk of abandonment with the multi-bundle treatment when taking credit for in-cabinet detection. The results of the sensitivity study are provided below.

Table 4-16 - Main Control Room Abandonment Bundle Treatment Sensitivity Delta CDF and Delta LERF Results

	Unit 1		Unit 2	
	ΔCDF [/yr]	ΔLERF [/yr]	ΔCDF [/yr]	ΔLERF [/yr]
VFDRs	1.2E-06	1.3E-08	2.7E-06	5.2E-09
Recovery Actions	1.0E-06	1.0E-07	1.0E-06	1.0E-07
Total	2.2E-06	1.1E-07	3.7E-06	1.1E-07

A manual summation may differ from the Total due to rounding in the last digit.

Table 4-17 - Main Control Room Abandonment Bundle Treatment Sensitivity Total CDF and Total LERF Results

	Unit 1		Unit 2	
	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]
Internal Events plus External Flooding and High Winds	1.4E-05	6.2E-07	1.4E-05	6.2E-07
Fire[1]	2.0E-05	4.3E-06	2.0E-05	4.0E-06
Fire – Recovery Actions[2]	1.2E-06	1.2E-07	1.2E-06	1.2E-07
Total	3.6E-05	5.0E-06	3.5E-05	4.8E-06

[1] Fire results credit operator actions to secure power during control room abandonment for loss of control sequences.
[2] Values are for recovery actions associated with control room abandonment due to environmental reasons.
A manual summation may differ from the Total due to rounding in the last digit.

4.8.3.11 Maintenance Influence Factor Sensitivity

Fire compartments that were assigned a maintenance influence factor of 10 were considered in this sensitivity for re-assignment to a factor of 50. A total of 5 fire compartments were selected. One fire compartment was selected for the Control/Reactor Building area for each unit, and one fire compartment was selected for the Turbine Building area for each unit. The single fire compartment selected for the Brunswick Wide area is common to both units. The results of the sensitivity study are provided below.

Table 4-18 – Maintenance Influence Factor Sensitivity Delta CDF and Delta LERF Results

	Unit 1		Unit 2	
	ΔCDF [/yr]	ΔLERF [/yr]	ΔCDF [/yr]	ΔLERF [/yr]
VFDRs	1.2E-06	1.3E-08	2.7E-06	5.2E-09
Recovery Actions	9.1E-07	9.1E-08	9.1E-07	9.1E-08
Total	2.1E-06	1.0E-07	3.6E-06	9.6E-08

A manual summation may differ from the Total due to rounding in the last digit.

Table 4-19 - Maintenance Influence Factor Sensitivity Total CDF and Total LERF Results

	Unit 1		Unit 2	
	CDF [/yr]	LERF [/yr]	CDF [/yr]	LERF [/yr]
Internal Events plus External Flooding and High Winds	1.4E-05	6.2E-07	1.4E-05	6.2E-07
Fire[1]	2.0E-05	4.3E-06	2.0E-05	4.0E-06
Fire – Recovery Actions[2]	1.0E-06	1.0E-07	1.0E-06	1.0E-07
Total	3.5E-05	5.0E-06	3.5E-05	4.8E-06

[1] Fire results credit operator actions to secure power during control room abandonment for loss of control sequences.
[2] Values are for recovery actions associated with control room abandonment due to environmental reasons.
A manual summation may differ from the Total due to rounding in the last digit.

5.0 REGULATORY EVALUATION

5.1 Introduction – 10 CFR 50.48

On July 16, 2004, the NRC amended 10 CFR 50.48, Fire Protection, to add a new subsection, 10 CFR 50.48(c), which establishes alternative fire protection requirements. 10 CFR 50.48 endorses, with exceptions, NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition (NFPA 805), as a voluntary alternative for demonstrating compliance with 10 CFR 50.48 Section (b), Appendix R, and Section (f), Decommissioning.

The voluntary adoption of 10 CFR 50.48(c) by BSEP does not eliminate the need to comply with 10 CFR 50.48(a) and 10 CFR 50, Appendix A, GDC 3, Fire Protection. The NRC addressed the overall adequacy of the regulations during the promulgation of 10 CFR 50.48(c) (Reference Federal Register (FR) Notice 69 FR 33536 dated June 16, 2004, ML041340086).

> *"NFPA 805 does not supersede the requirements of GDC 3, 10 CFR 50.48(a), or 10 CFR 50.48(f). Those regulatory requirements continue to apply to licensees that adopt NFPA 805. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements may be met is different than under 10 CFR 50.48(b). Specifically, whereas GDC 3 refers to SSCs important to safety, NFPA 805 identifies fire protection systems and features required to meet the Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is ensured is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805. The Section 1.5.1 criteria include provisions for ensuring that reactivity control, inventory and pressure control, decay heat removal, vital auxiliaries, and process monitoring are achieved and maintained.*
>
> *This methodology specifies a process to identify the fire protection systems and features required to achieve the nuclear safety performance criteria in Section 1.5 of NFPA 805. Once a determination has been made that a fire protection system or feature is required to achieve the performance criteria of Section 1.5, its design must meet any applicable requirements of NFPA 805, Chapter 3. Having identified the required fire protection systems and features, the licensee selects either a deterministic or performance-based approach to demonstrate that the performance criteria are satisfied. This process satisfies the GDC 3 requirement to design and locate SSCs important to safety to minimize the probability and effects of fires and explosions." (Reference FR Notice 69 FR 33536 dated June 16, 2004, ML041340086)*

The new rule provides actions that may be taken to establish compliance with 10 CFR 50.48(a), which requires each operating nuclear power plant to have a fire protection program plan that satisfies GDC 3, as well as specific requirements in that section. The transition process described in 10 CFR 50.48(c)(3)(ii) provides, in pertinent parts, that a licensee intending to adopt the new rule must, among other things, "modify the fire protection plan required by paragraph (a) of that section to reflect the licensee's decision to comply with NFPA 805." Therefore, to the extent that the contents of the existing fire protection program plan required by 10 CFR 50.48(a) are inconsistent with NFPA 805, the fire protection program plan must be modified to achieve compliance with the requirements in NFPA 805. All other requirements of 10 CFR 50.48(a) and GDC 3 have corresponding requirements in NFPA 805.

A comparison of the current requirements in Appendix R with the comparable requirements in Section 3 of NFPA 805 shows that the two sets of requirements are consistent in many respects. This was further clarified in FAQ 07-0032, 10 CFR 50.48(a), and GDC 3 clarification (ML081400292). The following tables provide a cross-reference of fire protection regulations associated with the post-transition BSEP fire protection program and applicable industry and BSEP documents that address the topic.

10 CFR 50.48(a)

Table 5-1 10 CFR 50.48(a) – Applicability/Compliance Reference

	10 CFR 50.48(a) Section(s)	Applicability/Compliance Reference
(1)	Each holder of an operating license issued under this part or a combined license issued under part 52 of this chapter must have a fire protection plan that satisfies Criterion 3 of appendix A to this part. This fire protection plan must:	See below
	(i) Describe the overall fire protection program for the facility;	NFPA 805 Section 3.2 NEI 04-02 Table B-1
	(ii) Identify the various positions within the licensee's organization that are responsible for the program;	NFPA 805 Section 3.2.2 NEI 04-02 Table B-1
	(iii) State the authorities that are delegated to each of these positions to implement those responsibilities; and	NFPA 805 Section 3.2.2 NEI 04-02 Table B-1
	(iv) Outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage.	NFPA 805 Section 2.7 and Chapters 3 and 4 NEI 04-02 B-1 and B-3 Tables
(2)	The plan must also describe specific features necessary to implement the program described in paragraph (a)(1) of this section such as:	See below
	(i) Administrative controls and personnel requirements for fire prevention and manual fire suppression activities;	NFPA 805 Sections 3.3.1 and 3.4 NEI 04-02 Table B-1
	(ii) Automatic and manually operated fire detection and suppression systems; and	NFPA 805 Sections 3.5 through 3.10 and Chapter 4 NEI 04-02 B-1 and B-3 Tables
	(iii) The means to limit fire damage to structures, systems, or components important to safety so that the capability to shut down the plant safely is ensured.	NFPA 805 Section 3.3 and Chapter 4 NEI 04-02 B-3 Table
(3)	The licensee shall retain the fire protection plan and each change to the plan as a record until the Commission terminates the reactor license. The licensee shall retain each superseded revision of the procedures for 3 years from the date it was superseded.	NFPA 805 Section 2.7.1.1 requires that documentation (Analyses, as defined by NFPA 805 2.4, performed to demonstrate compliance with this standard) be maintained for the life of the plant. RDC-NGGC-0001

Table 5-1 10 CFR 50.48(a) – Applicability/Compliance Reference

10 CFR 50.48(a) Section(s)	Applicability/Compliance Reference
(4) Each applicant for a design approval, design certification, or manufacturing license under part 52 of this chapter must have a description and analysis of the fire protection design features for the standard plant necessary to demonstrate compliance with Criterion 3 of appendix A to this part.	Not applicable. BSEP is licensed under 10 CFR 50.

General Design Criterion 3

Table 5-2 GDC 3 – Applicability/Compliance Reference

GDC 3, Fire Protection, Statement	Applicability/Compliance Reference
Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.	NFPA 805 Chapters 3 and 4 NEI 04-02 B-1 and B-3 Tables
Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room.	NFPA 805 Sections 3.3.2, 3.3.3, 3.3.4, 3.11.4 NEI 04-02 B-1 Table
Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety.	NFPA 805 Chapters 3 and 4 NEI 04-02 B-1 and B-3 Tables
Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components	NFPA 805 Sections 3.4 through 3.10 and 4.2.1 NEI 04-02 Table B-3

10 CFR 50.48(c)

Table 5-3 10 CFR 50.48(c) – Applicability/Compliance Reference

10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
(1) *Approval of incorporation by reference.* National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (NFPA 805), which is referenced in this section, was approved for incorporation by reference by the Director of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51.	General Information. NFPA 805 (2001 edition) is the edition used.
(2) Exceptions, modifications, and supplementation of NFPA 805. As used in this section, references to NFPA 805 are to the 2001 Edition, with the following exceptions, modifications, and supplementation:	General Information. NFPA 805 (2001 edition) is the edition used.
(i) *Life Safety Goal, Objectives, and Criteria.* The Life Safety Goal, Objectives, and Criteria of Chapter 1 are not endorsed.	The Life Safety Goal, Objectives, and Criteria of Chapter 1 of NFPA 805 are not part of the LAR.
(ii) *Plant Damage/Business Interruption Goal, Objectives, and Criteria.* The Plant Damage/Business Interruption Goal, Objectives, and Criteria of Chapter 1 are not endorsed.	The Plant Damage/Business Interruption Goal, Objectives, and Criteria of Chapter 1 of NFPA 805 are not part of the LAR.
(iii) *Use of feed-and-bleed.* In demonstrating compliance with the performance criteria of Sections 1.5.1(b) and (c), a high-pressure charging/injection pump coupled with the pressurizer power-operated relief valves (PORVs) as the sole fire-protected safe shutdown path for maintaining reactor coolant inventory, pressure control, and decay heat removal capability (i.e., feed-and-bleed) for pressurized-water reactors (PWRs) is not permitted.	BSEP is a BWR. This is not applicable.
(iv) Uncertainty analysis. An uncertainty analysis performed in accordance with Section 2.7.3.5 is not required to support deterministic approach calculations.	Uncertainty analysis was not performed for deterministic methodology.
(v) Existing cables. In lieu of installing cables meeting flame propagation tests as required by Section 3.3.5.3, a flame-retardant coating may be applied to the electric cables, or an automatic fixed fire suppression system may be installed to provide an equivalent level of protection. In addition, the italicized exception to Section 3.3.5.3 is not endorsed.	Electrical cable construction complies with a flame propagation test that was found acceptable to the NRC as documented in NEI 04-02 Table B-1.
(vi) Water supply and distribution. The italicized exception to Section 3.6.4 is not endorsed. Licensees who wish to use the exception to Section 3.6.4 must submit a request for a license amendment in accordance with paragraph (c)(2)(vii) of this section.	BSEP complies as documented in Attachment A. See NEI 04-02 Table B-1.

Table 5-3 10 CFR 50.48(c) – Applicability/Compliance Reference

	10 CFR 50.48(c) Section(s)	Applicability/Compliance Reference
	(vii) Performance-based methods. Notwithstanding the prohibition in Section 3.1 against the use of performance-based methods, the fire protection program elements and minimum design requirements of Chapter 3 may be subject to the performance-based methods permitted elsewhere in the standard. Licensees who wish to use performance-based methods for these fire protection program elements and minimum design requirements shall submit a request in the form of an application for license amendment under § 50.90. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the performance-based approach; (A) Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (B) Maintains safety margins; and (C) Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).	The use of performance-based methods for NFPA 805 Chapter 3 is requested. See Attachment L.
(3)	*Compliance with NFPA 805.*	See below
	(i) A licensee may maintain a fire protection program that complies with NFPA 805 as an alternative to complying with paragraph (b) of this section for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979. The licensee shall submit a request to comply with NFPA 805 in the form of an application for license amendment under § 50.90. The application must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases thereof. The Director of the Office of Nuclear Reactor Regulation, or a designee of the Director, may approve the application if the Director or designee determines that the licensee has identified orders, license conditions, and the technical specifications that must be revised or superseded, and that any necessary revisions are adequate. Any approval by the Director or the designee must be in the form of a license amendment approving the use of NFPA 805 together with any necessary revisions to the technical specifications.	The LAR was submitted in accordance with 10 CFR 50.90. The LAR included applicable license conditions, orders, technical specifications/bases that needed to be revised and/or superseded.
	(ii) The licensee shall complete its implementation of the methodology in Chapter 2 of NFPA 805 (including all required evaluations and analyses) and, upon completion, modify the fire protection plan required by paragraph (a) of this section to reflect the licensee's decision to comply with NFPA 805, before changing its fire protection program or nuclear power plant as permitted by NFPA 805.	The LAR and transition report summarize the evaluations and analyses performed in accordance with Chapter 2 of NFPA 805.
(4)	Risk-informed or performance-based alternatives to compliance with NFPA 805. A licensee may submit a request to use risk-informed or performance-based alternatives to compliance with NFPA 805. The request must be in the form of an application for license amendment under § 50.90 of this chapter. The Director of the Office of Nuclear Reactor Regulation, or designee of the Director, may approve the application if the Director or designee determines that the proposed alternatives: (i) Satisfy the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release; (ii) Maintain safety margins; and (iii) Maintain fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).	No risk-informed or performance-based alternatives to compliance with NFPA 805 (per 10 CFR 50.48(c)(4)) were utilized. See Attachment P.

5.2 Regulatory Topics

5.2.1 License Condition Changes

The current BSEP fire protection license condition 2.B.(6) is being replaced with the standard license condition based upon Regulatory Position 3.1 of RG 1.205, as shown in Attachment M.

5.2.2 Technical Specifications

BSEP conducted a review of the Technical Specifications to determine which Technical Specifications are required to be revised, deleted, or superseded. BSEP determined that the changes to the Technical Specifications and applicable justification listed in Attachment N are adequate for the BSEP adoption of the new fire protection licensing basis.

5.2.3 Orders and Exemptions

A review was conducted of the BSEP docketed correspondence to determine if there were any orders or exemptions that needed to be superseded or revised. A review was also performed to ensure that compliance with the physical protection requirements, security orders, and adherence to those commitments applicable to the plant are maintained. A discussion of affected orders and exemptions is included in Attachment O.

5.3 Regulatory Evaluations

5.3.1 No Significant Hazards Consideration

A written evaluation of the significant hazards consideration of a proposed license amendment is required by 10 CFR 50.92. According to 10 CFR 50.92, a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

- Involve a significant increase in the probability or consequences of an accident previously evaluated; or
- Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- Involve a significant reduction in a margin of safety.

This evaluation is contained in Attachment Q.

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. BSEP has evaluated the proposed amendment and determined that it involves no significant hazards consideration.

5.3.2 Environmental Consideration

Pursuant to 10 CFR 51.22(b), an evaluation of the LAR has been performed to determine whether it meets the criteria for categorical exclusion set forth in

10 CFR 51.22(c). That evaluation is discussed in Attachment R. The evaluation confirms that this LAR meets the criteria set forth in 10 CFR 51.22(c)(9) for categorical exclusion from the need for an environmental impact assessment or statement.

5.4 Revision to the UFSAR

After the approval of the LAR, in accordance with 10 CFR 50.71(e), the BSEP UFSAR will be revised. The content will be consistent with NEI 04-02.

5.5 Transition Implementation Schedule

The following schedule for transitioning BSEP to the new fire protection licensing basis requires NRC approval of the LAR in accordance with the following schedule:

- Implementation of new NFPA 805 fire protection program to include procedure changes, process updates, and training to affected plant personnel. This will occur 180 days after NRC approval. If the turnover is due to fall within an outage window then the changes will be implemented 60 days after startup from the scheduled outage.
- Modifications will be completed by the startup of the second refueling outage for each unit after issuance of the Safety Evaluation (SE). Appropriate compensatory measures will be maintained until modifications are complete.

6.0 REFERENCES

The following references were used in the development of the TR. Additional references are in the Attachments.

NRC Documents

1. Letter, NRC to NEI, Process for Frequently Asked Questions For Title 10 of The Code Of Federal Regulations, Part 50.48(c) Transitions, July 12, 2006 (ML061660105).
2. NRC Enforcement Policy, Policy Statement: Revision, Federal Register, Vol. 69, No. 115, June 16, 2004, pp. 33684–33685.
3. NRC Generic Letter 86-10, Supplement 1, Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area, March 25, 1994.
4. NRC Regulatory Issue Summary 2007-19: Communicating Clarifications of Staff Positions in RG 1.205 Concerning Issues Identified During Pilot Application of NFPA Std 805, August 20, 2007 (ML071590227).
5. NUREG/CR-6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, April 2005.
6. Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 1 – November 2002.
7. Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 2 - March 2009).
8. Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, Revision 1, December 2009.
9. Voluntary Fire Protection Requirement for Light-Water Reactors; Adoption of NFPA 805 as a Risk-Informed, Performance-Based Alternative, Final Rule, Federal Register, Vol. 69, No. 115, June 16, 2004, pp. 33536–33551.

Other Industry Documents

1. ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S 2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, American Society of Mechanical Engineers/American Nuclear Society, New York, NY.
2. EPRI Fire Protection Equipment Surveillance Optimization and Maintenance Guide TR-1006756, Final Report July 2003
3. NEI 00-01, Guidance for Post-Fire Safe Shutdown Circuit Analysis, Revision 1, January 2005.

4. NEI 00-01, Guidance for Post-Fire Safe Shutdown Circuit Analysis, Revision 2, May 2009.
5. NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c), Revision 2 April 2008.
6. NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition.

Licensee Correspondence

1. Letter CP&L to NRC, Letter of Intent to Transition to 10 CFR 40.48(c), June 10, 2005 (ML051720404)
2. Letter NRC to CP&L, Grants Enforcement Discretion Regarding NFPA Standard 805, April 29, 2007 (ML070590625).
3. Letter NRC to CP&L, Issuance of Amendment Regarding Adoption of NFPA Standard 805, Safety Evaluation for the Shearon Harris Nuclear Power Plant, June 28, 2012 (ML101130535)