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PNP 2013-079

October 24, 2013

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

SUBJECT: Response to Request for Additional Information – License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

References:

1. ENO letter, PNP 2012-106, "License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors," dated December 12, 2012 (ADAMS Accession Number ML12348A455)

2. ENO letter, PNP 2013-013, "Response to Clarification Request — License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors," dated February 21, 2013 (ADAMS Accession Number ML13079A090)

3. NRC electronic mail of August 8, 2013, "Palisades - Requests for Additional Information Regarding Transition to the Fire Protection Program to NFPA Standard 805 (TAC No. MF0382)" (ADAMS Accession Number ML13220B131)

4. ENO letter, PNP 2013-075, "Response to Request for Additional Information – License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors", dated September 30, 2013

Dear Sir or Madam:

In Reference 1, Entergy Nuclear Operations, Inc. (ENO) submitted a license amendment request to adopt the NFPA 805 performance-based standard for fire protection for light water reactors. In Reference 2, ENO responded to a clarification request. In Reference 3, ENO received electronic Request for Additional Information (RAIs). In Reference 4, ENO submitted the 60-Day RAI responses.

Attachment 1 provides the ENO responses to the 90-day RAIs, as follows:

Requests for Additional Information	**Response Time**	**Response Date**
SSA RAI 05, 07 PRA RAI 04, 06, 14, 15, 21, 22	90 Days	November 1, 2013

A copy of this response has been provided to the designated representative of the State of Michigan.

This letter contains no new or revised commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 24, 2013.

Sincerely,

ajv/jpm

Attachments: 1. Response to Request for Additional Information Regarding License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC
State of Michigan

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO ADOPT NFPA 805 PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTORS

Electronic RAIs were received from the Nuclear Regulatory Commission (NRC) on August 8, 2013. The Entergy Nuclear Operations, Inc. (ENO) responses to the 90-Day RAIs are provided below.

The ENO documents referenced in the RAI responses below are provided in the Palisades NFPA 805 LAR References Portal under the 90-Day RAI Response Reference Folder.

NRC Request

Safe Shutdown Analysis (SSA) RAI 05

LAR Section 4.2.1.2 addresses the safe and stable condition as hot shutdown. Provide a more detailed description of the systems, evolutions, and resources required to maintain this condition. Include the following items:

> *a) LAR Section 4.2.1.2, "Safe and Stable Conditions for the Plant, End State Characterization" states that the internal events model does not take credit for repairs, but it does not describe what is credited in the fire probabilistic risk assessment (FPRA). The LAR also states that "Consequences of failures that occur when hot shutdown is reached were examined to ensure that a safe and stable state is achieved. If a safe and stable state was not achieved, then an appropriate plant damage state was assigned." Provide clarification as to what "an appropriate plant damage state" means and how it is used. Describe whether repairs were required to establish a safe and stable condition for the FPRA. If so, explain what repairs were considered, and how they were included in the model.*
>
> *b) Provide a description of system capacity limitations and/or time-critical actions for systems (e.g., gas/air supply for control valves, boron supply, direct current (DC) battery power, diesel fuel) needed to maintain safe and stable conditions similar to what was provided for AFW system water supply (100,000 gallons/8 hours).*
>
> *c) Provide a more detailed qualitative description of the level of risk associated with the failure of operator actions and equipment necessary to sustain safe and stable conditions for an extended period of time.*

ENO Response

SSA RAI 05

NFPA-805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition”, describes safe and stable conditions as follows:

> “Safe and Stable Conditions. 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.”

NFPA-805 also provides requirements associated with established nuclear safety performance criteria stating:

> “Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met.
>
> (a) Reactivity Control. Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.
>
> (b) Inventory and Pressure Control. With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained for a PWR and shall be capable of maintaining or rapidly restoring reactor water level above top of active fuel for a BWR such that fuel clad damage as a result of a fire is prevented.
>
> (c) Decay Heat Removal. Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.
>
> (d) Vital Auxiliaries. Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.
>
> (e) Process Monitoring. Process monitoring shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.”

Palisades Nuclear Plant (PNP) LAR Attachment C, Table B-3, provides a summary discussion for each fire area as to the ability to meet the nuclear safety performance criteria and includes a discussion of the fire Probabilistic Risk Assessment (PRA) results. For a general overview refer to the section titled: Nuclear Safety Performance

Criteria – Method of Accomplishment provided for each fire area. Additional information supporting the ability to maintain NFPA 805 safe and stable is discussed below.

a) Plant Damage State

Plant damage state is defined as a group of accident sequence endstates that have similar characteristics with respect to accident progression, and containment or engineered safety feature operability according to plant conditions at the onset of severe core damage. The plant conditions considered are those that determine the capability of the containment to cope with a severe core damage accident. The plant damage states represent the interface between the Level 1 (identification and quantification of the sequences of events leading to the onset of core damage) and Level 2 (evaluation of containment response to severe accident challenges and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment) analyses.

An appropriate plant damage state is a generic phrase representing the correct assignment of accident sequence end state to plant damage state based on the specific set of system and equipment failures and availabilities that can impact the progression of a severe accident. Refer to the details contained in PNP's Notebook NB-PSA-SS for more information on safe and stable states.

Repairs

The nuclear safety capability assessment documented in PNP report PLP-RPT-12-00143, establishes one train of equipment that is free from fire damage and therefore, functional. This ensures the performance criteria, as outlined in NFPA 805, are met and the plant can be placed in a safe and stable condition. Recovery actions are credited to restore equipment functions. Repair of equipment is not credited in this assessment as being relied upon to achieve safe and stable conditions.

PNP LAR Section 4.2.1.2: The PNP full power internal events model does not take credit for repair of failed equipment. Reaching a non-core-damage end-state for the purpose of establishing a safe and stable state implies success of operator recovery actions without crediting repairs. For the FPRA, component repairs are not credited. Safe and stable conditions are achieved though available systems and components and by recovery actions that restore lost functions as established under the nuclear safety capability assessment.

b) The ability to supply resources such as water, fuel oil and nitrogen in achieving and maintaining NFPA 805 safe and stable conditions is important. Such resources are finite but plant design and established site inventories support the initial stages of system operation that may be time critical in establishing safe and stable conditions. Time critical actions at the system level place the plant in a safe configuration using these resources and can vary based on the nature and location of the postulated fire. The resources discussed below provide capability to support actions needed to achieve safe and stable conditions. Longer term, these resources are replenished as needed using onsite sources or off-site sources. Where needed, maintaining safe and stable may be resolved by switching to other resources (Lake Michigan for water needs) or mechanisms (manual operation) that support the required system function and the ability to maintain safe and stable conditions. Action to replenish supplies or switch to alternate capabilities are considered to parallel routine operating activities and therefore present a low risk to the plant in maintaining safe and stable conditions.

Air Supply Systems

Air Supply – Instrument and Service Air System

The instrument and service air supply has three 100% capacity air compressors that can be individually aligned to an operating diesel generator. Additional capacity is available from two 100% capacity feedwater purity air compressors. The instrument air system however is mostly constructed from soldered copper piping and as such is assumed to fail in all fire areas where an air supply header exists. Associated equipment, specifically air operated valves are designed to fail to a predetermined safe position. No recovery actions or repairs are established to recover the instrument and service air system except for a containment fire where the supply can be isolated from the control room should the header fail in this area.

For air controlled valves supporting safe shutdown conditions, selected valve controls have been supplied with backup nitrogen or compressed air supplies. These alternate supplies are connected to the valve's air supply line and isolated only with check valves. Should the air supply fail the alternate supply will align without the need for operator action. In cases where operation may be required and all air and nitrogen supplies are not available, the individual valve has manual operating capability either on the valve or through operation of a nearby manual valve.

The instrument and service air system as a whole is not credited for maintaining NFPA 805 safe and stable conditions. Reference PNP report PLP RPT-12-00143 for evaluation of the instrument and service air system in each fire area. Reference Design Basis Document DBD-1.05, for details associated with this system.

Air Supply – High Pressure Air

The high pressure air system is designed as two separate safety related trains, each supporting one train of safety related components. A third non-safety compressor supports non-safety loads. Each high pressure air compressor can fail or is removed from service under load shed conditions without immediate loss of function. Compressors maintain large accumulators associated with each train at minimum pressure providing the capability of stroking all associated valves twice in a Large Break (Loss of Coolant Accident (LOCA) scenario. For other postulated events that may result in a higher demand the compressors can be manually restored to their associated safety related power supply.

High pressure air is not credited for maintaining NFPA 805 safe and stable conditions. Reference PNP report PLP RPT-12-00143 for evaluation of the high pressure air systems.

Reference Design Basis Document DBD-1.05 for additional details associated with the high pressure air system.

Air Supply – Emergency Diesel Starting Air

Each emergency diesel generator has its own air supply system used exclusively for starting the emergency diesel generator. Two tanks support two 100 % capacity air start systems for each emergency diesel. The air storage tanks are maintained at minimum pressure that will allow two start attempts per train should the associated compressor fail. Should the tanks require filling, each train is automatically filled from a common air compressor that operates off a 480 VAC supply. If power is not available and the tank contents have been exhausted, the compressor can be driven from a gas power engine. Operating procedures are in place to operate the air supply systems for emergency diesel generator under normal and off-normal conditions.

Reference Technical Specification 3.8.3 and its associated bases document B3.8.3 Diesel Fuel, Lube Oil, and Starting Air. Additional details can be found in Design Basis Document DBD-5.01.

The emergency diesel starting air is not specifically evaluated in the NSCA. It is considered a subsystem of the emergency diesel generator.

Battery Capability

Two trains of station batteries are capable of supplying their associated 125 VDC Buses with power for 4 hours. Under postulated fire events, bus loading may increase as hot shorts may potentially increase load. To maintain battery life, recovery actions address disconnecting the battery from the associated 125 VDC bus through the use of a shunt trip allowing only one breaker panel to supply control power to critical loads such as the associated 2400 VAC safety bus loads and diesel generator equipment.

Panels EC-150/EC-150A remote hot shutdown monitoring and control panels are powered from one of the two 125 VDC buses. Recovery actions address use of the shunt trip to protect the battery associated with EC-150/EC-150A's power supply. Once the main 125 VDC bus has been removed from the battery, the battery has been shown to be able to supply EC-150/150A for at least 72 hours. Calculation EA-APR-95-035 provides the initial conditions and the evaluation demonstrating a battery life of up to 72 hours.

A modification to allow the connection of a remote generator to EC-150/EC-150A for extended operation beyond battery life is included in the modifications associated with the NFPA 805 Transition. Reference PNP LAR Attachment S, Table S-2, item S2-19.

For events where a battery may lose charging capability recover actions are established to cross connect a battery charger from the opposite train's ac power to restore the 125 VDC Bus power supply. Loss of a given battery charger may occur if operation is aligned in a cross connected configuration where one train of charging is connected to the opposite trains battery.

PNP LAR Attachment S, Table S-2, modification item S2-26 provides the capability to align a cross train battery charger to the same train power supply.

Reference PNP report PLP RPT-12-00143 for evaluation of the 125 VDC electrical system and fire areas where the shunt trip is used to protect battery life.

<u>Boric Acid/Inventory Supply for Primary System Inventory</u>

<u>T-58, Safety Injection Refueling Water Storage (SIRW) Tank</u>

Capacity/Contents: 285,000 gals of water > 1720 ppm Boron.

The Safety Injection Refueling Water (SIRW) tank Technical Specification minimum required inventory for operability during power operation is 250,000 gallons of borated water. In Mode 4 a minimum inventory of 200,000 gallons is required. Values are based on Technical Specification 3.5.4 and associated Bases Document B3.5.4. The technical specification bases document also notes that approximately 17,000 gallons of borated water is needed to bring the plant from hot standby to cold shutdown. Water needed to reach cold shutdown operation is based on shrinkage, normal flows from the primary system such as seal leak off or sampling and inventory exchange required to provide boric acid concentrations needed for shutdown conditions. Minimum boric acid concentration in the SIRW tank is required to be > 1720 ppm and < 2500 ppm boron. This concentration supports negative reactivity shutdown requirements under emergency and refueling conditions. The SIRW tank is the only borated water source credited to maintain NFPA 805 safe and stable conditions. Recovery actions to align the SIRW tank to a charging pump or to a high pressure safety injection pump are addressed in selected fire areas where control of isolation valves is lost. See PNP LAR Attachment G for actions that align this water supply. Charging pumps and high pressure safety injection pumps are capable of being supplied from this inventory.

Borated water addition is not critical in achieving safe and stable conditions. Reactivity conditions to achieve safe and stable are met by a reactor trip and all control rods inserted in the core. Reference details contained in PNP Notebook NB-PSA-SS for more information on reactivity and primary water system inventory needs.

T-53A and T-53B Boric Acid Storage Tanks

Capacity/Contents: ~6500 gallons of borated water in each tank.

The concentrated boric acid system includes two tanks, valves, pumps and piping that can provide additional flow paths (pumped or gravity feed) for concentrated boric acid from the storage tanks to the suction side of the charging pumps. System configuration is shown on drawing M-202 Sheet-1A. The system and components shown have not been credited in maintaining NFPA 805 safe and stable conditions. However, the key components that can supply inventory from these tanks have been evaluated in the Nuclear Capability Assessment for all fire areas. Equipment, including motor operated valves, remains available providing multiple paths to supply borated water to the suction of the charging pumps. Reference PNP reports PLP-RPT-12-00143 and PLP-RPT-12-00060.

Additional Non-credited Water Sources

Multiple non-credited water sources for maintaining primary system inventory are available. These additional sources are discussed in PNP LAR Section 4.2.1.2.

Nitrogen Supply – Backup to Instrument Air

Nitrogen Supply – AFW Steam Turbine Control Valve

The one auxiliary steam supply control valve controlling steam to the turbine driven auxiliary feedwater pump is provided with a backup nitrogen supply to the normal instrument air supply. The nitrogen supply is designed to support steam supply control valve operation upon loss of instrument air for extended period of time. The steam supply valve has manual operating capability should all air and nitrogen be lost.

Nitrogen Supply – AFW Flow Control Valves

The two auxiliary feedwater level control valves are provided with backup nitrogen to the normal instrument air supply. The nitrogen supply is designed to support the level control operation upon loss of instrument air for extended period of time. Both level control valves have manual operating capability should all air and nitrogen be lost.

Nitrogen Supply – Atmospheric Steam Dump Valves (ASDV)

The steam generator atmospheric steam dump valves are supplied from instrument air and have a backup nitrogen supply tied to the plant's bulk nitrogen system. The bulk nitrogen supply header can also be supplied from a set of nitrogen tanks that are normally isolated from the bulk nitrogen header providing additional capacity.

Loss of air supply will result in the atmospheric steam dump valves failing closed. In achieving safe and stable conditions venting of steam through the main steam relief valve is credited.

In addition to the ASDV air and nitrogen supplies discussed above, an additional control system and nitrogen supply is being added as part of the NFPA 805 Transition modifications. Reference PNP LAR Attachment S, Table S-2, item S2-12 for additional details.

Reference design bases document DBD-1.05 for information on nitrogen support systems.

Fire Water Supply

The three fire water pumps take suction from the Intake Structure basin which is supplied from Lake Michigan. This water source is considered to be an unlimited water supply for firefighting activities. This water supply also supports safety related flows such as the supply to the auxiliary feedwater pump suction, should condensate storage inventory be exhausted.

Fuel Oil Supply

On site fuel oil supply is maintained in multiple tanks. A fuel oil transfer system maintains all fuel oil tanks filled using T-10A as the source supply. PNP site procedures, along with necessary equipment, support manual transfer of fuel oil for emergency diesel and diesel fire water pump day tanks.

T-10A, Fuel Oil Storage Tank

Capacity 50,000 gallons of fuel oil.

Technical Specification 3.8.3 requires minimum inventory in the Fuel Oil Storage Tank and Diesel Generator Day Tank allowing one emergency diesel to operate for 7 days (Approximately 33,000 gallons) of fuel. The 7 day period is based on the emergency diesel operating at accident loading conditions. Lower loadings associated with NFPA 805 safe and stable conditions would extend operating ability beyond 7 days. Reference Technical Specification 3.8.3 and its associated bases document B3.8.3.

The fuel oil transfer system will automatically fill emergency diesel generator and diesel fire water day tanks from T-10A, Fuel Oil Storage Tank. Should the fuel oil transfer system and the associated auto fill capability fail, manual filling of various tanks can occur. The manual fill procedures and equipment are in place as discussed in PNP LAR Section 4.2.1.2. The diesel fuel inventory of T-10A over the long term is maintained through filling from off-site sources.

T-926, Feedwater Purity Fuel Oil Tank

Capacity: 30,000 gallons

Design capacity of this tank is 30,000 gallons of fuel oil. Inventory may vary based on plant needs. Available inventory is used for manually filling the emergency diesel generator day tanks using an established procedure to fill the fuel oil day tanks from this source. Equipment for manually filling the day tanks is staged. The manual fill procedures and equipment are in place as discussed in PNP LAR Section 4.2.1.2. Inventory of this tank can also be maintained through filling from off-site sources.

T-25A or T-25B Diesel Generator Day Tanks

Capacity: 3355 gallons per day tank

Each diesel generator's day tank is required by Technical Specifications to contain at least 2500 gallons of fuel oil. This volume of fuel oil is sufficient fuel for approximately 13.5 hours of full load operation before the tank must be refilled from onsite storage. Reference Technical Specification 3.8.3 and its associated bases document B3.8.3. When the normal fuel oil transfer system is not available, manual fill procedures are in place as discussed in PNP LAR Section 4.2.1.2. Recovery actions to manually fill a diesel generator fuel oil day tank using T-926, Feedwater Purity Fuel Oil Tank or T-10A, Fuel Oil Storage Tank as the supply inventory, are in place for fire areas where a diesel generator is required to maintain NFPA 805 safe and stable conditions.

Diesel Generator Lubricating Oil

The onsite storage of diesel generator lubricating oil is sufficient to ensure 7 days of continuous operation. This supply is sufficient to allow replenishing lubricating oil from offsite sources. Reference Technical Specification 3.8.3 and its associated bases document B3.8.3.

T-24 and T-40 Diesel Fire Water Pump Day Tanks

Capacity: 275 gallons

Each diesel fire pump day tanks provides several hours of pumping operation from the low level fill point in the tank. This will support fire protection loads during the course of a postulated fire. Automatic fill occurs as part of the normal operation of the fuel oil transfer system. Should the fuel oil transfer system, along with the auto fill capability be lost, manual fill procedures are in place to transfer fuel oil from T-10A, Fuel Oil Storage Tank to either T-24 or T-40 diesel fire pump fuel oil day tanks.

c) Operator actions are established in the nuclear safe capability assessment with respect to equipment required to achieve and maintain NFPA 805 safe and stable conditions. PNP Report PLP RPT-12-00143 addresses equipment failure and actions that may be required for recovery. The equipment and associated operator actions are then evaluated in the fire PRA establishing the risk associated with achieving safe and stable conditions.

Once safe and stable conditions have been achieved, maintaining these conditions parallels actions that are similar to plant shutdown. Site emergency organizations, as well as off-site resources will be aligned to support evaluation, planning and performance of ongoing operating activities needed to maintain the plant in a safe condition. Operation under these conditions presents a low risk environment for maintaining safe and stable conditions as activities have shifted from immediate time critical response actions to an evaluated and planned state of operation.

REFERENCES

- NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants”, 2001 Edition,
- Drawing M-202 Sht-1A, “Piping and Instrument Diagram- Chemical and Volume Control System” Revision 63
- NB-PSA-SS, “Palisades Safe and Stable States”, Revision 2
- PLP-RPT-12-00143, “Nuclear Safety Capability Assessment Fire Area Analysis Results” Revision 0
- PLP-RPT-12-00060, “IN-92-18 Motor Operated Valve Failure Evaluation” Revision 0
- Technical Specification 3.5.4, “Safety Injection Refueling Water Tank (SIRWT)”, Amendment 189
- Technical Specification 3.8.1, “AC Sources – Operating”, Amendment 219
- Technical Specification 3.8.3, “Diesel Fuel, Lube Oil, and Starting Air” Amendment 242
- Technical Specification Bases 3.5.4, “Safety Injection Refueling Water Tank (SIRWT)” Amendment 227
- Technical Specification Bases B 3.8.1, “AC Sources – Operating”, Revision Date 11/08/12
- Technical Specifications Bases B 3.8.3, “Diesel Fuel, Lube Oil, and Starting Air”, Revision Date 9/16/2011
- Technical Specifications Bases B 3.8.4, “DC Sources – Operating”, Revision Date 7-13-06
- EA-APR-95-035, "10CFR50 Appendix R Alternate Shutdown Battery Capacity Analysis”, Revision 1
- Design Basis Document DBD-5.01, “Diesel Engine and Auxiliary Systems”, Revision 6
- Design Basis Document DBD-1.05, “Compressed Air Systems”, Revision 5

NRC Request

SSA RAI 07

In LAR Attachment G, Table G-2, for Fire Area 23, the staff noted that a recovery action may be required to provide portable fans for cooling to the CR based on a postulated fire in Fire Area 22, turbine lube oil room. During the audit, the licensee indicated the portable fans used for this recovery action are gasoline powered.

The requirements of General Design Criterion 3 (GDC-3) state for fire protection that structures, systems, and components (SSCs) important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and CR. 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. 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.

The use of gasoline near the CR does not align with GDC-3. The use and refueling of a portable gasoline-powered blower presents a hazard to equipment important to nuclear safety. The use of portable fuel-fired equipment should be consistent with the requirements of GDC-3.

Provide an approach to resolving the subject VFDRs and providing CR ventilation that is consistent with the requirements of GDC-3.

ENO Response

SSA RAI 07

The introduction to Appendix A to Part 50 – General Design Criteria for Nuclear Power Plants states, “The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.”

This statement is understood to indicate that the General Design Criteria (GDC) applies to the design, fabrication, construction, testing, and performance requirements for permanent plant structures, systems, and components important to safety. This statement is not understood to apply to temporary or portable equipment used to mitigate the impact of events such as a fire.

There are activities that periodically exceed the normal combustible loading of an area and the use of temporary or portable equipment would fit into this category. If a plant was restricted to never exceeding the in-situ combustibles, then it would be impractical to complete some activities needed to maintain and operate the facility safely.

The PNP response to GDC-3 is included in the Final Safety Analysis Report (FSAR) and states, "This criterion is met by designing the plant so that buildings containing critical portions of the plant such as the containment building, control room and auxiliary building are constructed of noncombustible, flame retardant and heat resistant materials. Plant areas critical for a safe shutdown have been divided into fire areas such that a fire in any given area will not propagate to other areas and will not impair the Plant's ability to safely shut down.

Through a series of modifications including installation of fire stops, cable separation, addition of sprinklers, addition of designated fire brigade, procedure changes and others, the Palisades Plant has established conformance to the requirements of 10 CFR 50.48 and Appendix R."

GDC-3 is related to permanent plant design and as described in the FSAR, the plant was designed and modified to establish conformance with 10 CFR 50.48 and Appendix R. The portable gasoline engine driven blowers are temporary equipment used to cope with the effects of a fire and are strategically located within the plant for use by the fire brigade and for use as temporary ventilation. These fans are staged with empty gas tanks and limited amounts of gasoline are appropriately staged within flammable storage cabinets within the plant. Neither the staged fans nor the flammable storage cabinets are located next to the control room. The gasoline is staged and transported as needed in safety cans for flammable liquids. The flammable storage cabinets and safety cans are design features to minimize the probability and effect of fires and explosions. The gasoline engine driven blowers are periodically tested so as to maintain them in a ready condition. These particular gasoline engine driven blowers are manufactured and marketed specifically as firefighting equipment.

Per PNP System Operating Procedure 24 (SOP-24), the configuration used for temporary ventilation places the gasoline engine driven blowers to pull air through an area, such as the control room, rather than blow air into an area. This configuration prevents exhaust fumes from being pushed into the area of concern. Another benefit of the gasoline engine driven blowers is that they are essentially self-contained which simplifies the process of putting temporary ventilation in place in a timely manner. Other types of alternative fans would require some type of power supply, which would delay establishment of temporary ventilation needed to address the effects of a fire. Compared to other alternative fans, the simple nature of a gas powered fan provides a high degree of flexibility, very low resources to establish needed ventilation and an approach that would not challenge resources at a time when focus must be on safe operation of the plant.

If the use and refueling of a portable gas-powered blower presented a hazard similar to the hazard of portable fuel-fired heaters, then it would have been identified and captured in NFPA 805 consistent with the prohibition on portable fuel-fired heaters, reference NFPA 805 Section 3.3.1.3.4. Since it was not captured in NFPA 805 similar to portable fuel-fired heaters, then it is understood that other fuel or gas powered devices were not considered to provide the same level of hazard and did not require a total prohibition.

Based on the above, PNP respectfully concludes that the use of gasoline powered blowers to cope with the effects of a fire is a more reliable technique as compared to other options, is consistent with the requirements of GDC-3, does not represent a significant hazard to the safe operation of the facility, and therefore maintains a higher margin of safety than other options.

REFERENCES

- Palisades Plant Final Safety Analysis Report (FSAR), Revision 30
- SOP-24, “Ventilation and Air Conditioning System”, Revision 61
- NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants”, 2001 Edition

NRC Request

PRA RAI 04

The ASME/ANS PRA Standard and RG 1.200, Rev. 2, provide guidance for the technical adequacy, including supporting requirements and peer reviews. Section 2.4.3.3 of NFPA 805 states that the PSA approach, methods, and data shall be acceptable to the AHJ. RG 1.205, provides guidance for use in complying with the requirements promulgated for risk-informed, performance-based fire protection programs that meet the requirements of 10 CFR 50.48(c) and the referenced 2001 Edition of NFPA 805. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Rev. 2, as providing methods acceptable to the NRC for adopting a fire protection program consistent with NFPA-805. The following additional information is requested in order for the staff to complete its review:

Based on Table 3 of Attachment 1 to the LAR Supplement dated February 21, 2013, PRM-14 has not been identified as being peer reviewed. The evaluation provided for PRM-B14 states that no new LERF phenomena are applicable to the FPRA that were not addressed for LERF estimation in the IEPRA. Discuss the process utilized or analysis performed to arrive at this conclusion.

ENO Response

PRA RAI 04

<u>Full Power Internal Events (FPIE) Large Early Release Frequency (LERF) Background</u>

In support of the 1993 Individual Plant Evaluation (IPE) Level 2 analysis, PNP performed a variety of plant-specific, detailed, and deterministic evaluations to evaluate LERF, instead of relying on the results of previous generic Probabilistic Safety Analyses (PSAs). Examples of these analyses included;

- Containment ultimate strength structural evaluation,
- Reactor cavity impulse load structural analysis,

- Thermal creep induced rupture of the primary coolant system (PCS) pressure boundary analyses
- Cavity pressurization during debris dispersal

Subsequent to the IPE, the Severe Accident Mitigation Alternatives (SAMA) analysis was performed in support of the license renewal project.

As a result, an assessment of accident phenomena and progression was conducted to review the methods used to quantify the IPE Containment Event Trees (CET). The "state of knowledge improvement" in several areas of severe accident phenomenology was reviewed. Examples of this assessment included high pressure melt ejection, direct containment heating, and induced hot leg rupture.

Current LERF Modeling

The current FPIE and LERF analysis follows the "Simplified Level 2 Modeling Guidelines," WCAP-16341-P, which many in the industry are currently using as a basis for updated Level 2 analyses.

This WCAP provides a common, standardized method for Pressurized Water Reactors (PWRs) with large dry containments to produce an analysis that generally meets capability category II of the American Society of Mechanical Engineers (ASME) Probabilistic Risk Assessment (PRA) standard. The guidance particularly addresses the latest understanding for induced steam generator tube ruptures, direct containment heating, and other important Level 2 phenomena. While the WCAP is focused on modeling LERF for the ASME standard, it includes guidance for intact, small, and late releases to provide a more complete, though still standardized, Level 2 analysis.

The WCAP provides an event tree structure for both station-blackout-related scenarios and non-station-blackout scenarios to determine the likelihood of different accident progression scenarios. Each event tree starts with a plant damage state and continues with questions that must be answered on a plant-specific basis. The WCAP provides straightforward guidance for the process.

During development of the Level 1 event trees applied to the fire PRA, it was concluded that all core damage sequences resulting from fire initiating events would end in an existing accident class and could be binned to the already defined plant damage states applied in the FPIE Level 2 PRA. As no new plant damage states were defined for the fire PRA, the existing containment phenomenology used in the Level 2 analysis is applicable to all fire initiating events.

The WCAP adaptation comports to the past detailed PNP IPE and SAMA employed LERF methods.

The detailed containment analysis performed in support of the PNP IPE and SAMA initiatives, and the current adaptation of the WCAP Level 2 methodology provides assurance that the PNP Fire LERF analysis is complete and that no new LERF phenomena are applicable to the FPRA that were not addressed for LERF estimation in the PNP FPIE PRA.

REFERENCES

- CP (Consumers Power) 1993. Palisades Plant – Individual Plant Examination for Severe Accident Vulnerabilities (IPE). Letter from Consumers Power Company to U.S. Nuclear Regulatory Commission. F341/1523. January 29
- NMC letter to NRC, "Application for Renewed Operating License," dated March 22, 2005
- Westinghouse Simplified Level 2 Modeling Guidelines – WOG Project: PA-RMSC-0088, WCAP-16341-P, Rev. 0, November 2005

NRC Request

PRA RAI 06

Per Section 11.5.1.6 of NUREG/CR-6850, transient fires should at a minimum be placed in locations within the plant PAUs where conditional core damage probabilities (CCDPs) are highest for that PAU, i.e., at "pinch points". Pinch points include locations of redundant trains or the vicinity of other potentially risk-relevant equipment, including the cabling associated with each. Transient fires should be placed at all appropriate locations in a PAU where they can threaten pinch points. Hot work should be assumed to occur in locations where hot work is a possibility, even if improbable, keeping in mind the same philosophy. Describe how transient and hot work fires are distributed within all PAUs (including the MCR). In particular, identify the criteria used to determine where an ignition source is placed within the PAUs. Also, if there are areas within a PAU where no transient or hot work fires are postulated because those areas are considered inaccessible, describe the criteria used to define "inaccessible." Note that an inaccessible area is not the same as a location where placement of a transient is simply unlikely. If there are "inaccessible" locations where hot work or transient fires are improbable and these locations are pinch points, provide a sensitivity study to determine the possible risk increase reflecting the possible size and frequency of fires in these locations.

ENO Response

PRA RAI 06

This RAI response provides an overview of the treatment of general transients and transients due to welding and cutting (hot work fires) in the PNP fire PRA. These fires are jointly referred to as transient fires.

Twenty-two (22) PAUs were analyzed with bounding, full-PAU burn-up scenarios such that specific transient fire scenarios were not needed. For the remaining eighteen (18) PAUs (which includes the MCR), transient fires were postulated in locations in which a 98^{th} percentile (317 kW) transient fire could damage a set of targets not otherwise already captured by a fixed ignition source. Using this process, pinch points are captured as a subset of fire scenario target sets.

Neither physical inaccessibility to a location nor improbability of a transient ignition source were used as specific criteria for development of transient fire scenarios within a PAU. However, upon review of transient fire treatments in response to this RAI, locations were noted where PRA cables were present, but transient scenarios were not postulated. Transient fire scenarios are being developed for these locations to demonstrate no pinch points were omitted and will be included in the base case results of the RAI Response fire PRA model presented in PRA RAI 23. Therefore, a sensitivity study to determine the possible risk increase reflecting the possible size and frequency of fires in these locations is not necessary.

NRC Request

PRA RAI 14

The Bin 15.2 ignition frequency from EPRI 1016735, "Fire PRA Methods Enhancements: Additions, Clarifications, and Refinements to EPRI 1011989," was further subdivided into frequencies associated with low- and medium-voltage panels as proposed by FAQ 06-0017, "Clarifying/Enhancing Guidance for Counting High Energy Arcing Faults in NUREG/CR-6850," (ADAMS Accession No. ML072500300, closure memo). Given that this FAQ was closed out prior to issuance of Supplement 1 to NUREG/CR-6850 (i.e., FAQ 08-0048), discuss the basis for frequencies 16a and 16b in Table 4-1 of the Plant Partitioning and Fire Ignition Frequency Development report, further justifying the approach provided in Attachment G.

ENO Response

PRA RAI 14

NUREG/CR-6850 provided the ignition frequency for electrical panel High Energy Arcing Faults (HEAF) as bin 16, without consideration to panel voltage. Later, FAQ 06-0017, partitioned this frequency into separate bins into low voltage panels (bin 16a) and medium voltage panels (bin 16b). Similarly, EPRI 1016735 provided the updated ignition frequency for electrical panel High Energy Arcing Faults (HEAF) as bin 15.2, without consideration to panel voltage. The EPRI data was later published in Supplement 1 to NUREG/CR-6850 as closeout of FAQ 08-0048. In order to provide consistent HEAF frequency bins for the original (NUREG) and updated (EPRI) data, and to preserve the refinement of the frequencies in terms of panel voltage, an analysis of the updated data was performed. This analysis, using the EPRI 1016735 data methodology, divided bin 15.2 into low voltage and medium voltage HEAFs consistent with FAQ 06-0017.

EPRI 1016735 Table B-4, Fire Ignition Bin Adjusted Counts and Associated Reactor Years, provided the following data for Bin 15.2, for a total of 2.5 events.

BIN 15.2 EVENTS

	1968-1990		1991-2000	
BIN	COUNTS	RX YEARS	COUNTS	RX YEARS
15.2	1.5	1376.2	1.0	1075.3

Based on the description of the HEAF events provided in NUREG/CR-6850 Appendix M, Table M-1, the counts were assigned to low voltage HEAF (bin 16a) and medium voltage HEAF (bin 16b) consistent with the bin numbering developed as a part of FAQ 06-0017:

BIN 15.2 EVENTS SPLIT INTO LOW / MEDIUM VOLTAGE

	1968-1990		1991-2000	
NEW BIN	COUNTS	RX YEARS	COUNTS	RX YEARS
16a	0.5	1376.2	0	1075.3
16b	1.0	1376.2	1.0	1075.3

These data were used to develop revised fire frequencies for low and medium voltage HEAFs using the statistical method outlined in EPRI 1016735, Section A.2, as this was the method used to develop the revised frequency for HEAF bin 15.2 provided in NUREG CR/6850 Supplement 1. This approach, which employs a constrained non-informative prior (CNIP) with a shaping factor of 0.5, the distribution parameters are calculated as follows:

Alpha_{prior}	= [count for 1968-1990] + 0.5
Beta_{prior}	= [1968-1990 reactor years]
Prior Mean	= Alpha_{prior} / Beta_{prior}
CNIP Prior Mean Beta	= 0.5 / Prior Mean
$\text{Alpha}_{posterior}$	= [count for 1991-2000] + 0.5
$\text{Beta}_{posterior}$	= CNIP Prior Mean Beta + [1991-2000 reactor years]
Posterior Mean	= $\text{Alpha}_{posterior}$ / $\text{Beta}_{posterior}$

The table below presents the distribution parameters (posterior) provided for the electrical panel HEAF bin Supplement 1 to NUREG/CR-6850 and the refined bins used as generic EPRI frequencies in the PNP fire PRA.

ELECTRICAL PANEL HEAF BIN DISTRIBUTION PARAMETERS

NUREG CR/6850 SUPPLEMENT 1				PALISADES FIRE PRA			
BIN	MEAN	ALPHA	BETA	BIN	MEAN	ALPHA	BETA
15.2	1.06E-3	1.5	1419	16a	2.84E-4	0.5	1763
				16b	9.78E-4	1.5	1534

In summary, the above establishes the basis for EPRI generic frequency bins 16a and 16b in Table 4-1 of the Plant Partitioning and Fire Ignition Frequency Development report.

REFERENCES

- FAQ 06-0017, “Clarifying/Enhancing Guidance for Counting High Energy Arcing Faults in NUREG/CR-6850,” (ADAMS Accession No. ML072500300, closure memo), September 26, 2007.
- FAQ 08-0048, “Revised Fire Ignition Frequencies,” (ADAMS Accession No. ML092180383, closure memo), September 1, 2009.
- EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, EPRI 1011089 -NUREG/CR-6850, August 2005.
- EPRI 1016735, Fire PRA Methods Enhancements: Additions, Clarifications, and Refinements to EPRI 1011989, December, 2008.
- EPRI 1019259, Fire Probabilistic Risk Assessment Methods Enhancements: Supplement 1 to NUREG/CR-6850 and EPRI 1011989. EPRI, Palo Alto, CA, and NRC, Washington, D.C.: December 2009.

NRC Request

PRA RAI 15

The ASME/ANS PRA Standard and RG 1.200, Rev. 2, provide guidance for the technical adequacy, including supporting requirements and peer reviews. Section 2.4.3.3 of NFPA 805 states that the PSA approach, methods, and data shall be acceptable to the AHJ. RG 1.205, provides guidance for use in complying with the requirements promulgated for risk-informed, performance-based fire protection programs that meet the requirements of 10 CFR 50.48(c) and the referenced 2001 Edition of NFPA 805. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Rev. 2, as providing methods acceptable to the NRC for adopting a fire protection program consistent with NFPA-805. The following additional information is requested in order for the staff to complete its review:

Provide additional justification for not postulating smoke damage. Address in this justification the specific types of components vulnerable to smoke damage and the *potential damage mechanisms presented in Appendix T of NUREG/CR-6850. Include discussion of the potential for smoke to cause failures in a common enclosure (e.g., bust ducts).*

ENO Response

PRA RAI 15

NUREG/CR-6850, Appendix T, Section T.2 identifies four modes of smoke damage: circuit bridging, contact fouling, binding of mechanical movement, and direct chemical/corrosive attack. Of these, only circuit bridging was found to be of potential risk significance. Exposure time plays a key role in the likelihood of failures from smoke; short term smoke damage will only result from a severe smoke exposure condition. Low voltage components, in particular instrument and control components, and higher voltage power components were identified as potentially susceptible to circuit bridging faults resulting from airborne and deposited smoke.

In the case of the Control Room, only abandonment scenarios would result in smoke exposure conditions sufficient to have a negative impact on components not directly impacted by fire damage. In these scenarios, no credit for instruments or controls potentially damaged by smoke was taken in the fire PRA. Similarly, no credit for surviving sensitive electronics was taken for those scenarios where the cabinet containing the sensitive electronics was also the ignition source. The walk-through control cabinets are provided with forced ventilation which further limits the potential for smoke damage.

External to the Control Room, low voltage instrumentation and control devices associated with credited equipment are housed within substantial panels. The panels, whether vented or unvented, prevent severe smoke damage to the equipment located within them from external fires. A fire within an enclosure was assumed to cause a loss of function of all equipment in the enclosures and therefore the effects of any smoke generated internally are bounded. No “exposed” sensitive electronics were identified during scenario walk downs.

High voltage power components are also contained within enclosures that would limit smoke density exposure. As with low voltage panels, a fire originating at a location within a power distribution enclosure (MCC, load center, transformer, or distribution panel) was assumed to cause a loss of function of the equipment, thus bounding the effect of smoke damage within the enclosure, including the potential impact on electrical bus bars. Segmented bus duct is only present in the outdoor yard area at PNP, and is therefore not subjected to concentrations of smoke produced by fires that do not directly fail the bus duct.

NRC Request

PRA RAI 21

The ASME/ANS PRA Standard and RG 1.200, Rev. 2, provide guidance for the technical adequacy, including supporting requirements and peer reviews. Section 2.4.3.3 of NFPA 805 states that the PSA approach, methods, and data shall be acceptable to the AHJ. RG 1.205, provides guidance for use in complying with the requirements promulgated for risk-informed, performance-based fire protection programs that meet the requirements of 10 CFR 50.48(c) and the referenced 2001 Edition of NFPA 805. RG 1.205 identifies NUREG/CR-6850 as documenting a methodology for conducting a FPRA and endorses, with exceptions and clarifications, NEI 04-02, Rev. 2, as providing methods acceptable to the NRC for adopting a fire protection program consistent with NFPA-805. The following additional information is requested in order for the staff to complete its review:

According to EA-PSA-FPIE-FIRE-12-04, SAPHIRE performs the fault tree and sequence quantification for the IEPRA using fault trees initially created in computer aided fault tree analysis (CAFTA). The final peer review indicates that the FPRA is quantified using systems analysis programs for hands-on integrated reliability evaluations (SAPHIRE), CAFTA and fracture analysis code (FRANC); however, it is unclear how the inputs and outputs of these codes are linked. Describe the quantification process utilized by the FPRA to support the LAR, and discuss to what extent this process and the resulting model have been peer-reviewed. Additionally, provide an overview of efforts performed to validate the model conversion documented in Appendix A of the Model Development report.

ENO Response

PRA RAI 21

The baseline FPIE logic model is normally quantified within the SAPHIRE framework. For the fire PRA model, the applicable portions of the CDF and LERF SAPHIRE sequence logic were converted to the CAFTA 1-TOP logic environment to facilitate use of the FRANC fire PRA software. Along with the fire PRA documentation, the fire PRA Peer Review team was provided with the CAFTA 1-TOP model, Fire PRA Database, FRANC model, and the quantification results of the fire PRA, from which cutsets were generated using the FTREX quantification engine.

<u>Quantification Process</u>

The quantification process used to support the PNP LAR is based on using CAFTA, FRANC, and FTREX. The CAFTA model contains the accident sequence logic. The conversion from SAPHIRE is discussed below.

The FRANC model contains fire scenario data including fire ignition frequencies (IGFs), non-suppression probabilities (NSPs), severity factors (SFs), mapping to failed basic events, and a list of basic events to be altered to specific values. The CCDP and CLERP for each fire scenario was calculated by applying the scenario specific basic event changes (setting basic events to TRUE or altering basic event probabilities) to the CAFTA logic model. Quantification followed with FTREX. The CDF and LERF values for each scenario were calculated within FRANC by multiplying the IGF, NSP, and SF terms by the CCDP/CLERP values. The overall CDF and LERF results are the sum of the individual fire scenario results.

The quantification process used to calculated delta risk calculations to support the VFDR analysis were performed in the same manner with additional adjustments made to calculate the potential risk reduction, if the VFDR was addressed. To calculate the risk reduction, the basic events associated with the VFDR were kept at their random failure probability instead of being set to TRUE or altered as specified in the base fire PRA calculations.

Migration of SAPHIRE logic model into CAFTA

The flow chart below provides a high level framework of the process;

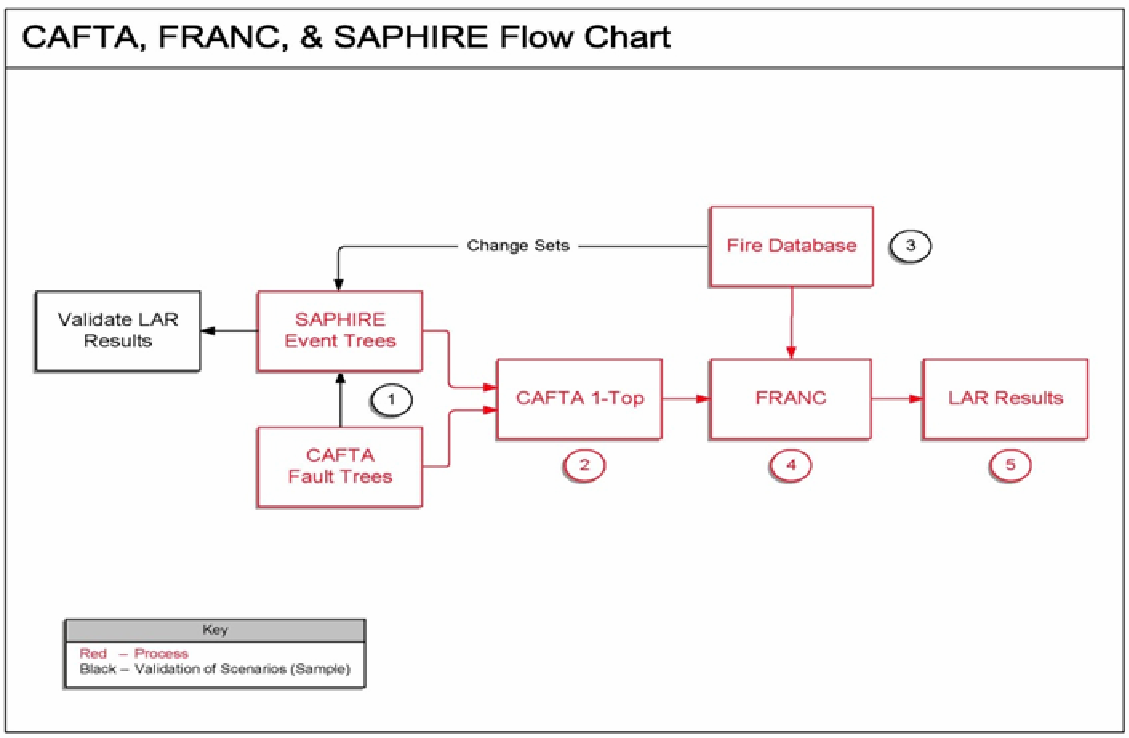

Figure 1: Flowchart of Model Conversion and Quantification

1. The multi-top CAFTA parent fault tree {1} (refer to the flowchart) provides the deductive logic, fault tree(s), for the SAPHIRE inductive logic modeling, event trees.
2. The event tree equations were extracted from SAPHIRE, modified to work with CAFTA and then input to PRAQuant. This task did not involve a full extraction of all sequences, but rather involved a limited scope conversion of the accident sequences required for use as part of the fire PRA model. That is, the conversion involved the “transient with main condenser available” event tree (TR-MCND-AV) sequences, and included conditional logic with

supporting fault trees. For example, all relevant event tree transfers (e.g. consequential ISLOCA, consequential LOCAs, and ATWS) were included.

3. Once the sequence cutsets and data were validated to match, PRAQuant was employed to create the CAFTA 1-Top {2}.
4. Again the CAFTA cutsets, in this case the results of quantifying the CAFTA 1-Top {2} were compared to the SAPHIRE single top set of results.
5. Given validation, the PRAQuant generated CAFTA 1-Top {2}, the SAPHIRE flag set data, and the project rules data (*.fay or *.mex) were subsequently provided to FRANC {4}.
6. The FRANC file {4} was populated with the appropriate scenario specific parameters (IGF, NSP, SF, affected components, and basic event alterations) using the data contained with the fire modeling database {3}
7. Coupled with the CAFTA 1-Top model and the scenario specific parameters, FRANC {4} sent the scenario specific model to FTREX for quantification. FTREX generated the fire CCDP and CLERP results that were loaded back into FRANC.
8. The CDF and LERF values calculated in FRANC (product of IGF, NSP, SF, and CCDP/CLERP) were used as inputs to the LAR {5}.
9. Finally, the fire database {3} was also used to generate SAPHIRE "change sets" data for a given PAU. SAPHIRE cutsets were then generated using the "ones" data to compare to the FRANC "ones" results for selected scenarios. This step was only used as an independent check of a sampling of the FRANC results and was not directly used in the compilation of results for the LAR.

Validation of SAPHIRE to CAFTA conversion

The process to validate the conversion of the transient with main condenser available sequences, as well as any transfers to other event trees was performed in two parts;

1. The first part involved validation of the individual sequence results obtained using CAFTA, PRAQuant, and FTREX vs. those obtained with SAPHIRE. This is step 3 in the process outlined in the Figure. The cutsets obtained from the two different quantification methods were compared at the basic event level.
2. After the sequence validation is complete, the CAFTA 1-Top model was created using PRAQuant. The results of solving the CAFTA 1-Top logic were then compared to the SAPHIRE endstate results using the same process as the sequence validation.

In summary, the above describes an overview of the FPRA conversion and quantification process as well as an outline of the elements reviewed by the peer review.

NRC Request

PRA RAI 22

SR DA-D9 does not appear to have been assigned a CC by the peer review for the IEPRA. Confirm that the peer review for the IEPRA and FPRA considered the clarifications and qualifications from RG 1.200, Rev. 2, March 2009 to the ASME/ANS PRA Standard.

ENO Response

PRA RAI 22

Both the Internal Events Probabilistic Risk Assessment (IEPRA) and Fire Probabilistic Risk Assessment (FPRA) peer review teams considered the clarifications and qualifications from Regulatory Guide (RG) 1.200 Rev. 2 dated March 2009.

Section 1.2 of the final IEPRA peer review report states:

In July 2009, Entergy contracted with Westinghouse to perform a full scope PRA peer review of the Palisades PRAs to determine compliance with Addendum A of the ASME/ANS Combined PRA standard and RG 1.200 R2. This review was conducted under the auspices of the Pressurized Water Reactor Owners Group task PA-RMSC-0386R1. This report documents the results of this PRA peer review for Palisades.

Section 1.3 of the final FPRA peer review report states:

The specific technical items and criteria for assignment of capability categories are based on checklists developed directly from the Standard High Level Requirements (HLRs) and Supporting Requirements (SRs). These checklists also include the resolutions in R.G. 1.200, Revision 2, including those which were not included directly in the Standard and inquiries on the Standard and Frequently Asked Questions (FAQs) as appropriate.

The peer review teams documented findings and observations using an electronic database. The database fields repeated Supporting Requirement (SR) Capability Category (CC) I, II, and III text from the American Society of Mechanical Engineers/ American Nuclear Society (ASME/ANS) standard as well as RG 1.200 resolution with the standard. The database did not contain a record for supporting requirement DA-D9; this requirement only appears in RG 1.200 and does not appear in the ASME/ANS standard.

An excerpt from Table A-2 of RG 1.200 for SR DA-D9 is shown below to illustrate that this was a new requirement from the NRC staff and not a requirement described in the ASME standard.

Excerpt from Table A-2 RG 1.200 Rev. 2			
Index No	**Issue**	**Position**	**Resolution**
DA-D9	New requirement needed, DA-C15 was incomplete, only provided for data collection, not quantification of repair. (See SY-A24.)	Qualification	Cat I, II, and III: For each SSC for which repair is to be modeled, ESTIMATE, based on the data collected in DA-C15, the probability of failure to repair the SSC in time to prevent core damage as a function of the accident sequence in which the SSC failure appears.

PNP does not model equipment repairs in the PRA, therefore, this requirement is not applicable and has no impact on the model used in the NFPA 805 PNP LAR submittal.

REFERENCES

- FPIE-PEER-LTR-RAM-II-10-015, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements For The Palisades Nuclear Power Plant Probabilistic Risk Assessment"
- Regulatory Guide 1.200 Rev. 2, "An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities"
- 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"
- FIRE-PRA-PEER-17825-1, "Palisades Fire PRA Peer Review to Requirements in Part 4 of the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications (Report 17825-1)"