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UNITED STATES NUCLEAR REGULATORY COMMISSION

Site-Specific Simulation of Offsite Emergency Response for SOARCA

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Presentation to ACRS

December 7, 2006

EP Modeling

- Modeling the protective response afforded by NPP Emergency Preparedness (EP) programs substantially improves realism
- All NPPs have regularly inspected and exercised EP programs
- Modeling realistically represents NRC Defense-in-Depth Policy

ASSUMPTIONS

- Officials will implement emergency plans
- The public will largely obey direction from officials
- Emergency workers will implement the plans
- Basis from NUREG/CR-6864, “Identification and Analysis of Factors Affecting Emergency Evacuations” and PAR Study Focus Groups

Emergency Declaration

- Emergencies will be declared when EALs are reached
- Control room readings not available to SOARCA project, but can be inferred from MELCOR output
- "SRO discretion EAL" may be considered

Precautionary Actions

- Early precautionary actions are taken at Alert and Site Area Emergency
- Evacuation of special needs populations
  - Schools
  - Parks
- Prepare nursing homes
- Sirens sound and the public is notified
  - Shadow evacuation

Population Movement

- Evacuation Time Estimates (ETEs) provide:
  - Site-specific evacuation travel times
  - Population preparation time
- Divide population into cohorts
- Cohorts start at different times and move at different speeds

MACCS2 is being modified to accommodate multiple cohorts

Time of Day

- Accounting for variations in cohort travel for time of day, time of year, weather, peak population densities, etc. goes beyond current scope/resources
- A composite estimate for each cohort will consider these variations
  - Assumptions documented

Travel Speed

- Limited access roads and towns affect evacuation speed
  - Reflected in cohort travel speed where practical
- **MACCS is being modified to allow variation of travel speed by cohort in space and time**

Beyond the EPZ

- Protective actions beyond the EPZ are required by regulation but detailed planning is not
- Need would be identified via dose projection (plant, state, NRC) but implementation is ad hoc
- Population density, scenario timing, road networks and shadow evacuation will inform estimates of public preparation time and evacuation speed
- Less detailed than within EPZ

Radial Evacuation

- MACCS2 models radial evacuation
- Evacuation routes are not radially outward
- MACCS2 has been modified to easily model lateral movement
  - Improves realism
- Travel speed will be estimated for each cohort and modified by roads and towns

KI

- Considered for programs that use it
- For pre-distributed KI assume 50% of the population takes it
- For programs that do not use KI, 0% will be assumed
- Where KI is distributed at congregate care centers (and the like), 20% assumed
- Assumptions used for all cohorts

ISSUES

- Assumptions made regarding discretionary protective action decisions by offsite response organizations (OROs)
  - Develop ORO advisory group
- Some ETEs are very old
  - Develop models based on best available information

ISSUES

- Probabilistic representation of weather affects modeling of evacuation
  - Estimate cohort speeds as though one quadrant were evacuated
- MACCS2 run time for latent cancer fatality threshold calculations is affected by number of cohorts
  - Minimize evacuation cohorts (e.g., some leave before release)

Going Forward

- Industry looks forward to working with NRC and DHS:
  - Staff discussions on NUREG 0800 Section 13.3
  - Review and approval of NEI 07-01

NEI

State-of-the-Art Reactor Consequence Analyses ACRS Meeting

December 7, 2006

Robert J. Prato
Office of Nuclear Regulatory Research

AGENDA

- MELCOR AND MACCS CODE IMPROVEMENTS
- PLANT GROUPING
- SCENARIO SELECTION
- LNT – vs – THRESHOLD
- EMERGENCY PREPAREDNESS
- ACRS ISSUES AND QUESTIONS

CODE IMPROVEMENTS

- 4 of 4 MELCORE CODE IMPROVEMENTS ARE BEING IMPLEMENTED
- 8 OF 10 MACCS2 CODE IMPROVEMENTS ARE BEING IMPLEMENTED
- 2 MACCS2 CODE IMPROVEMENTS ARE NOT BEING IMPLEMENTED
  - WET DEPOSITION MODEL AEROSOL SIZE DEPENDENCE
  - ANGULAR RESOLUTION

PLANT GROUPINGS

- GE, Mark 1
- GE, Mark 2
- GE, Mark3
- B&W, Dry Ambient
- CE, Dry Ambient
- W, 4 loop, Ice Condenser
- W, 2 and 3 loop, Dry Ambient, and Dry Sub-atmospheric
- W, 4 loop, Dry Ambient, and Dry Sub-atmospheric

USE OF CDF / RELEASE FREQUENCY

FULL-SCOPE LEVEL -2 PRAs ARE NOT AVAILABLE FOR ALL PLANTS, LIMITING THE STAFF'S ABILITY TO SELECT SCENARIOS BASED ON RELEASE FREQUENCY.

FOR THE PURPOSE OF SOAR-CA, THE NRC IS CONSIDERING DEFINING "RELEASE" BROADLY AS EARLY OR LATE, LARGE OR SMALL. ON THE BASIS THIS DEFINITION, ALL CORE DAMAGE EVENTS WILL RESULT IN A RELEASE

HENCE, THE STAFF IS EVALUATING SCENARIOS SELECTION USING CORE DAMAGE FREQUENCY.

Selection of Scenarios to Use for Consequence Analysis

Current PRA Tools

MODEL/METH[illegible]	[illegible] PLANTS [illegible] MODELS
INTERNAL EVENTS	
SPAR Models	103
EXTERNAL EVENTS	
EE SPAR Models	13
IPEEE- Seismic PRAs	37
IPEEE- Seismic Margin Analysis	66
IPEEE- Fire PRA	23
IPEEE- FIVE Methodology (FIVE+, FIVE/PRA, and FIVE/FPRAIG)	80

SCENARIO SELECTION OPTIONS

- INTERNAL EVENTS CDF WITH UNCERTAINTY CONSIDERATIONS
- INTERNAL EVENTS CDF WITH UNCERTAINTY AND EXTERNAL EVENTS CONSIDERED

INTERNAL EVENTS CDF WITH UNCERTAINTY

- USE SPAR CDF FACTORING IN UNCERTAINTY, EXCLUDE EXTERNAL EVENTS, TO DETERMINE SCENARIO SELECTION

- IMPLEMENT USING INDIVIDUAL PLANT RESULTS OR SELECT DOMINANT SCENARIOS FOR CLASS OF PLANT

- NOT VIABLE, BETTER OPTIONS AVAILABLE
  - SIMPLISTIC APPROACH
  - EXCLUDES EFFECTS OF EXTERNAL EVENTS

INTERNAL EVENTS CDF WITH UNCERTAINTY AND EXTERNAL EVENTS CONSIDERED

- USE SPAR CDF FACTORING IN UNCERTAINTY AND EXTERNAL EVENTS TO DETERMINE SCENARIO SELECTION

- EXTERNAL EVENTS CAN BE INCLUDED USING OLD DATA OR NEW DATA (SCENARIOS and CDFs) OBTAINED FROM LICENSEES. WHERE NEW DATA NOT AVAILABLE, CONSIDER USING MEAN VALUES

- IMPLEMENT USING INDIVIDUAL PLANT RESULTS OR SELECT DOMINANT SCENARIOS FOR CLASS OF PLANT

- VIABLE OPTIONS AVAILABLE
  - BEST APPROACH FOR INCLUDING EXTERNAL EVENTS
  - SIMPLISTIC APPROACH FOR PLANTS WITH NO EXTERNAL EVENTS PRAs

LNT – vs – THRESHOLD

- The Commission directed the staff not solely rely on conservative collective dose models to assess latent cancer health effects from low doses of radiation, but to utilize a range of potential latent cancer health effects estimated from low levels of radiation.

- The staff identified a range of thresholds from 0 to 5 rem.

- To use a range of 0 to 5 rem, would require the use of Linear, no threshold for the treatment of "0" dose in modeling, and for the remaining range of doses would require a threshold.

- Options for Doses, the staff is considering the use of 0, 100 mrem, 1 rem, and 5 rem

- The staff is considering different methods of presenting the results that we will be prepared to present at the next ACRS meeting.

EMERGENCY PREPAREDNESS

Site-Specific Simulation
Of Offsite Emergency Response
for SOARCA

ACRS ISSUES AND QUESTIONS

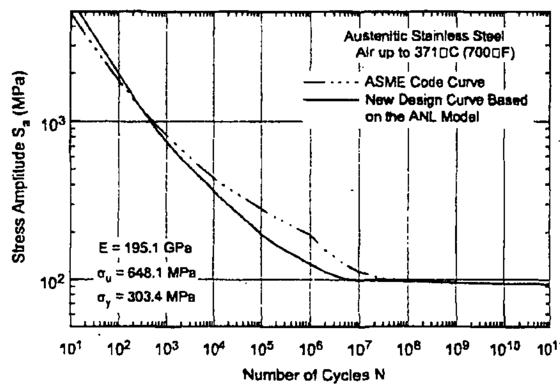
Regulatory Position 3: Ni-Cr-Fe Alloys (e.g., Alloy 600 and 690)

- ✓ Calculate fatigue usage in air with ASME Code Analysis procedures +
  - ✓ **New ANL model air SS curve**
- ✓ Calculate the F_{en} using
  - ✓ Equation A.14

    $$F_{en} = \exp(T^* O^* \dot{\varepsilon}^*)$$

    (Appendix A of NUREG/CR-6909)
- ✓ Calculate the environmental fatigue usage (U_{en})

$$U_{en} = U_1 F_{en,1} + U_2 F_{en,2} \ldots U_n F_{en,n}$$



Summary

- RG 1.207 endorses the use of new air curve for SSs
- RG 1.207 endorses the F_{en} methodology
- Guidance on incorporating environmental correction factor to fatigue design analyses
  - Appendix A of NUREG/CR-6909 Rev. 1
- NUREG/CR-6909 Rev. 1 describes in detail the technical basis

Resolution of Public Comments

- July 24, 2006 - DG-1144 and draft NUREG/CR-6909 published for public comments (60 day comment period)

- Public comment period ended September 25, 2006

Resolution of Public Comments (cont.)

- 8 correspondents submitted a total of 56 comments on DG-1144 and draft NUREG/CR-6909
  - All comments addressed individually

- Final RG 1.207 and NUREG/CR-6909 Rev. 1 reflects the resolution of these comments

- 6 main issues identified

Resolution of Public Comments (cont.)

Staff Response to Public Comments on DG-1144 and Draft NUREG/CR-6909

#	Source*	Comment**	Response
1	I - 1a.	Each Comment appears individually in this column.	NRC staff response for each comment.

*.

Source I	Ronnie L. Gardner, AREVA NP, Inc	[illegible]
Source II	Takao NAKAMURA, The Kansai Electric Power Co., Inc.	ML062790141
Source III	James H. Riley, Nuclear Energy Institute	ML062790134
Source IV	C. L. Funderburk, Dominion Resources Services, Inc	ML062790144
Source V	Makoto HIGUCHI, Ishikawajima-Harima Heavy Industries Co., Ltd	ML062790138
Source VI	Robert E. Brown, GE Energy Nuclear	ML062790142
Source VII	Gerry C. Slagis, G.C. Slagis Associates, Consulting Engineering	ML062630349
Source VIII	Kenneth R. Balkey, Nuclear Codes and Standards, American Society of Mechanical Engineers	ML062790139

**. Comments are quoted directly from the letter submitted by the commenter

Resolution of Public Comments (cont.)

- Six issues (comment id #'s):
  1. Operating experience and applicability of specimen data (1, 7, 14, 16, 45)
  2. Details on approach (22, 24, 27, 37)
  3. Ni-Cr-Fe alloy fatigue curve (20, 25, 44)
  4. Burden due to increase in locations required to be analyzed (2, 43)
  5. Overly conservative position (4, 5, 15)
  6. ASME Code case (56)

1. Operating experience and applicability of specimen data (1, 7, 14, 16, 45)

Issue:

- *There is no operating experience that supports the need for these conservative design rules.*
- *Comments questioning the applicability of specimen data being representative of actual components in service.*

Staff Response:

- Numerous examples of fatigue cracking of nuclear power plant components reported - EPRI TR-106696.
- Applicability of laboratory data to component behavior has been demonstrated by mock–up and component tests (references provided in previous presentation). In fact, is the basis for the current ASME Code fatigue curves.

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2. Details on approach (22, 24, 27, 37)

Issues:

- *References made to other guidance containing similar Fen approach (Japan) also acceptable/endorsed?*
- *"Since DG-1144 utilizes a similar Fen methodology to that evaluated in MRP-47, Rev.1, the issues identified in MRP-47, Rev.1 are considered to be equally applicable to the DG-1144 methodology. Some, but not all, of the issues raised in MRP-47, Rev.1 have been specifically addressed in DG-1144. Based on this, the MRP would like to see clarification on the remaining issues included in DG-1144 or the supporting document".*

Staff Response:

- The papers listed in NUREG/CR-6909 are for reference only. Section C, Regulatory Position, of the regulatory guide contains the methodology endorsed by the staff.
- The level of analytical detail discussed on additional items on MRP-47, Rev.1 are beyond the scope of this regulatory guide.

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3. Ni-Cr-Fe alloy fatigue curve (20, 25, 44)

Issue:

Provide guidance for Ni-Cr-Fe alloys (e.g., Alloy 600 and 690).

Staff Response:

The staff incorporated F_{en} methodology for Ni-Cr-Fe alloy materials into RG 1.207 (RP 3) and NUREG/CR-6909 Rev. 1 (Section 6).

4. Burden due to increase in locations required to be analyzed (2, 43)

Issue:

Increase in the CUFs will lead to more analyzed piping break locations, to more installed pipe whip restraints, and to designs that will be more detrimental for normal (thermal expansion) operating conditions.

Staff Response:

- Staff will consider a justified modification with the appropriate technical basis of the fatigue criteria for postulation of pipe breaks if implementation of the current criteria results in a significant increase in the number of required pipe whip restraints.

- The necessity for additional pipe restraints will disappear with a successful LBB analysis

5. Overly conservative position (4, 5, 15)

Issue:

Commenter believes that the alternative methods for fatigue analysis provided in NUREG/CR-6909 and DG-1144 are too conservative and should not be used for the design of new reactors.

Staff Response:

The staff position is based on a 95% confidence that there is less than 5% probability of fatigue crack initiation. Implementation of this criteria resulted in a carbon steel and low-alloy steel air curves which are less conservative than the existing ASME Code curve

6. ASME Code case (56)

Issue:

"ASME will continue to develop other Code Cases covering alternative ways of addressing [the impact of the LWR environment]... and the Code Case will be issued early in 2007. Once these Code Cases are issued, ASME requests the NRC to endorse these Code Cases in a revision of the Regulatory Guide 1.84".

Staff Response:

The NRC staff will consider endorsing available ASME Code Cases through its normal process for revising Regulatory Guide 1.84.

Revisions made from DG-1144 to RG 1.207

Main revision:

- The staff incorporated F_{en} methodology for Ni-Cr-Fe alloy materials into RG 1.207 (RP 3) and NUREG/CR-6909 Rev. 1 (Section 6).
- High Cycle Fatigue Regime (> 10^6 cycles)

Other:

Some editorial changes for clarification on the technical basisNUREG/CR-6909

Conclusion

RG 1.207 is ready for issuance

- Final RG 1.207 and NUREG/CR-6909 Rev. 1 reflects the resolution of these comments
- Final RG 1.207 and NUREG/CR-6909 Rev.1 will be published by March 2007 (High priority RG)
- Seeking ACRS concurrence to publish final effective guide

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PROVIDED FOR INTERNAL ACRS USE ONLY

State of the Art Reactor Consequence Analysis Information Request

Emergency Preparedness Information

Information request from all nuclear power plants. This will not change with class of plant.

1. Full Evacuation Time Estimate, not just a summary.

2. From the emergency plan implementing procedure state the procedure or preferably the operator aid, used for (and perhaps titled) Classification of Emergencies

3. From the State and County(ies) emergency plans:

   - The chapter or procedure (perhaps titled "Protective Actions",) that is used by decision makers to decide on the appropriate public protective actions during actual emergencies.

   - The chapters, appendices or procedures that address protective actions for special needs populations and protective actions for schools and when those actions should be considered or taken.

4. Full size color evacuation route map and evacuation travel direction for the public in the various emergency planning areas.

Human Reliability Analysis Information

Information request from all nuclear power plants. This will not change with class of plant.

1. Procedures - EOPs, SAMGs, & EDMGs, preferably in electronic form.
2. Plant staff contacts cognizant of operations, training, procedures, etc. (in order to understand procedure implementation).

Note: For "reference plants" only, a plant site visits needs to be scheduled to better understand procedure implementation & likelihood of operator success/failure.

Structural Information

Information request from all nuclear power plants. This will not change with class of plant.

PWR Plants:

Concrete Containments

1. Containment Building Liner Plate Drawings
2. Containment Building Reinforcement Details Drawings
3. Containment Building Equipment Hatch Drawings, including information about the bolt torques
4. Containment Building Personnel Airlock Drawing and material of the seals

Questions?

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Steel Containments

1. Steel containment drawing showing thickness, material, and overall dimensions (detailed fabrication drawings not needed).
2. Shield building reinforcement details drawings.
3. Containment Building Equipment Hatch Drawings, including information about the bolt torques.
4. Containment Building Personnel Airlock Drawing and material of the seals.

BWR plants:

1. Containment vessel and containment vessel head (drywell head) drawings
2. Suppression chamber drawings showing the thickness, material, and overall dimensions of the Torus. Detailed fabrication and support drawings for the torus are not required.
3. Provide, number, diameter, and magnitude of torque/tension for the bolts drywell head bolts as specified in the site procedure.
4. Location and diameter of the drywell/wetwell hardened vent, and the primary containment pressure limit (PCPL).

External Events Information

Information request from all nuclear power plants. This will not change with class of plant.

1. Have you updated you IPEEE since initial submittal to NRC
2. If so, provide the following:

   a. the date of the last update
   b. a list of the External Events sequence / scenario that have a CDF ≥1E-7
   c. provide a descriptions of each sequence / scenario

   Examples of the type of information we are looking for include: (1) the type of external event (e.g., fire, flood, seismic) that causes the sequence/scenario, (2) the subsequent initiating event (e.g., reactor trip, LOMFW, SLOCA, LLOCA), safety equipment and systems that are assumed to be unavailable for the sequence/scenario, and the CDF for the sequence/scenario.

MELCOR Information

PWR Westinghouse 4-Loop Large, Dry Lead Plants:

Reference Plant

General Plant Data

1. Current UFSAR (electronic preferred)
2. Estimated duration of station batteries during total ac power loss (with

and without load shedding if known)

3. Heat losses (typ) from reactor vessel and RCS piping.
4. 100% operating conditions (flows, temperatures, levels, pressures and pressure drops around RCS, make-up flow and temperature, etc.)
5. *Thermal-hydraulic model of RCS, containment, and auxiliary building*
6. Plant-specific MAAP or thermal-hydraulic input deck and supporting documentation
7. Depending on the availability and quality of this model, plant staff contacts cognizant of its MAAP model is being requested.

Reactor Vessel, RCS and SG data

1. Vessel cross-section drawings, including reactor internals
2. Weights and material type of reactor internal structures
3. Hot leg nozzle, surge line, and U-tube materials, dimensions, and drawings
4. Geometric drawings of the RCS piping, pressurizer, pressurizer relief tank, and steam lines
5. Type (composition), thickness, and thermal properties (e.g., k, ÿc_p) of vessel, RCS, and steam generator insulation
6. Provide design details regarding steam generator construction and performance (especially for Model F, if used)[1]

   - Model type
   - Primary and secondary water volume, inlet and outlet plenum volume, secondary steam volume
   - Tube diameter and number (number plugged)
   - Normal operating conditions (e.g., water and steam mass, water level, recirculation ratio, feedwater temperature, blowdown flow)
   - Geometric drawings (ideally with breakdown of volumes in each region)
   - Summary of internal structures (i.e., ideally component masses, surface area, and material construction) for tube sheet, separators, dryers, etc.

Core Data

1. Core fuel report and loading map for current cycle
2. Fuel data for 3 consecutive cycles

   - Fuel type (Vendor/model)
   - Total cycle burn-up
   - Cycle power history and shutdown time
   - Fuel loading pattern
   - Fuel assembly enrichment and MTU
   - Fuel assembly average power
   - BOC and EOC burn-ups for the assemblies

[1] We have information for Series 51 steam generator (i.e., used in several of the lead PWR units) but almost no information on the Series 44, Model D, and Model F generators (i.e., used in the remaining PWR units).

- o Average boron concentration
- o Average axial power profile (global or assembly-specific)
- o Peak fuel pin factor (optional)

Containment Data

1. Layout and vertical cross-section drawings of the containment
2. Summary of internal construction details for accident response model
    - o Compartment volumes
    - o Wall, floor, and ceiling thickness, surface area
    - o Connecting flow areas through doors, hatches, penetrations
    - o Estimate of miscellaneous metal mass/surface area
3. Water level calculation for pool depths during accident conditions
    - o Wet or dry reactor cavity for LOCAs, station blackout, etc.
    - o Under what conditions (if any) could the vessel lower head be flooded
4. Illustrations or description(s) of water drainage pathway(s) to basement and/or sump(s)
5. Best-estimate leak rate
6. Concrete chemical composition and rebar content

Auxiliary Building Data (for scenarios with release from containment to the adjacent auxiliary building)

1. Drawings detailing
    - o Room layout and dimensions
    - o Room connectivity/flow paths (doorways, stairwells, hatches)
    - o Ventilation system (design operation, filtering, dampers)
    - o Leakage rate %vol/day

Basic Trip and Actuation Logic Data

1. Reactor Protection System (SCRAM)
2. ECCS
3. Containment engineering safety systems (Spray, fan coolers, hydrogen recombiners)
4. Containment spray/ECCS RWST to recirculation mode switchover logic

Equipment Data

1. Auxiliary feedwater flow (esp., turbine-driven? power sources are required? control? rated flowrate?
2. ECCS description (pump-head curves), accumulators, and charging pumps

3. Containment spray flow curves
4. Fan cooler performance and description
5. RWST volume
6. Containment spray and ECCS heat exchanger performance

IPE or Later Plant Specific Severe Accident Insights

1. Pump-seal leakage characteristics following loss-of-seal cooling flow
2. Pressurizer and SG valve failure characteristics at saturated and high temperature conditions
3. Containment failure characteristics (pressure, temperature, location, and leakage rate or hole sizes)
4. Data for interfacing RHR LOCA (piping pathway and failure location, Aux. bldg. flooding characteristics, flooding consequences to other vital equipment, leak flowrate, etc.)
5. Unique plant-specific features that should considered in the severe accident analysis.[2]

Other PWR Non-reference Plants

Comparative and sensitive analyses will be performed with the lead plants for comparison to the reference plant results. Examples of sensitivity studies might include variations in steam generator response (Series 44, Series 51, Model D, and Model F), variations in dc battery life, or variations in auxiliary building design.

General Plant Data

1. Current UFSAR (electronic preferred).
2. Estimated duration of station batteries during total ac power loss (with and without load shedding if known)
3. Heat losses (typ) from reactor vessel and RCS piping.
4. 100% operating conditions (flows, temperatures, levels, pressures and pressure drops around RCS, make-up flow and temperature, etc.)

Thermal-hydraulic Model of RCS, Containment, and Auxiliary Building Data

1. Plant-specific MAAP or thermal-hydraulic input deck
2. Depending on the availability and quality of this model, many other requests may be unneeded.

Reactor Vessel, RCS, and SG Data

1. Type (composition), thickness, and thermal properties (e.g., k, ρc_p) of vessel, RCS, and steam generator insulation
2. Provide design details regarding steam generator construction and

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2 In a previous NRC mixed-oxide research projects, one of the candidate plants could align a water source from a nearby lake above the plant directly into the reactor vessel for low-pressure flooding and/or post-vessel failure containment flooding,.

performance (especially for Model F, if used)[3]

- o Model type
- o Primary and secondary water volume, inlet and outlet plenum volume, secondary steam volume
- o Tube diameter and number (number plugged)
- o Normal operating conditions (e.g., water and steam mass, water level, recirculation ratio, feedwater temperature, blowdown flow)

Core Information Data

1. Fuel type (Vendor/model)
2. Total cycle burn-up
3. Fuel loading map for current cycle
4. Assembly power, enrichment, BOC burn-up

Containment Data

1. Layout and vertical cross-section drawings of the containment
2. Total volume and compartment volumes
3. Water level calculation for pool depths during accident conditions
    - o Wet or dry cavity for LOCAs, station blackout, etc.
    - o Under what conditions (if any) could the vessel lower head be flooded
4. Illustrations or description(s) of water drainage pathway(s) to basement and/or sump(s)
5. Best-estimate leak rate
6. Concrete chemical composition and rebar content

Auxiliary Building Data (for scenarios with release from containment to the adjacent auxiliary building)

1. Drawings detailing
    - o Room layout and dimensions
    - o Room connectivity/flow paths (doorways, stairwells, hatches)
    - o Ventilation system (design operation, filtering, dampers)
    - o Leakage rate %vol/day

Switchover Logic

1. RWST volume
2. Containment spray/ECCS RWST to recirculation mode switchover logic

Equipment Data

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3 We have information for Series 51 steam generator (i.e., used in several of the lead PWR units) but almost no information on the Series 44, Model D, and Model F generators (i.e., used in the remaining PWR units).

1. Auxiliary feedwater flow (esp., turbine-driven? power sources are required? control? rated flowrate?
2. Fan cooler performance and description

IPE or later plant specific severe accident insights

1. Pump-seal leakage characteristics following loss-of-seal cooling flow
2. Pressurizer and SG valve failure characteristics at saturated and high temperature conditions
3. Containment failure characteristics (pressure, temperature, location, and leakage rate or hole sizes)
4. Data for interfacing RHR LOCA (piping pathway and failure location, Aux. bldg. flooding characteristics, flooding consequences to other vital equipment, leak flowrate, etc.)
5. Unique plant-specific features that should considered in the severe accident analysis.

Sequoyah

PWR fuel data for decay heat and fission product inventories are available from a previous NRC high fuel burn-up research project. In particular, the core loading, decay heat, and fission product inventory for Sequoyah will be used as a surrogate for all the large 4-loop PWR plants (i.e., with thermal power scaling). Sequoyah uses Framatome 17x17 Alliance fuel whereas the other lead PWR plants use Westinghouse 17x17 Vantage fuel. The data from the reference and lead plants plus the data from Sequoyah will be used to analyze any differences before proceeding. The following request to Sequoyah updates the previous project's data request and confirms the current fuel configuration (i.e., relative to the application from the previous high-burn-up project).

Fuel data for 3 consecutive cycles

- o Fuel type (Vendor/model)
- o Total cycle burn-up
- o Cycle power history and shutdown time
- o Fuel loading pattern
- o Fuel assembly enrichment and MTU
- o Fuel assembly average power
- o BOC and EOC burn-ups
- o Boron concentration
- o Average axial power profile (global or assembly-specific)
- o Peak fuel pin factor (optional) [4]

MACCS INFORMATION

1. Existing MACCS meteorological file - OR - one year from the on-site

[4] In a previous NRC mixed-oxide research projects, one of the candidate plants could align a water source from a nearby lake above the plant directly into the reactor vessel for low-pressure flooding and/or post-vessel failure containment flooding..

meteorological tower.

2. Precipitation data if available (which they are not required to gather) - OR- we need to know that they do not measure precipitation data.

3. Estimates of mixing height, by season and by day vs night - OR- we need to know that they do not this data.

DRAFT

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Plant/Containment Class Matrices

BWRs

MELCOR Class	Class 1			Class 2		Class 3
	GE 2 / Mark 1	GE 3 / Mark 1	GE 4 / Mark 1	GE 4 / Mark 2	GE 5 / Mark 2	GE 6 / Mark 3
1	Nine Mile Point 1	Dresden 2	Browns Ferry 2	Limerick 1	Columbia	Clinton
2	Oyster Creek	Dresden 3	Browns Ferry 3	Limerick 2	LaSalle 1	Grand Gulf
4		Monticello	Brunswick 1	Susquehanna 1	LaSalle 2	Perry
5		Pilgrim	Brunswick 2	Susquehanna 2	Nine Mile Point 2	River Bend
6		Quad Cities 1	Cooper			
7		Quad Cities 2	Duane Arnold			
8			Hatch 1			
9			Hatch 2			
10			Fermi 2			
11			Hope Creek			
12			Fitzpatrick			
13			Peach Bottom 2			
14			Peach Bottom 3			
15			Vermont Yankee			
	2	6	14	4	4	4
	22			8		4

Total # of BWRs = 34

PWRs

MELCOR Class	Class 4	Class 5	Class 6	Class 7*			Class 8	
	B&W / Dry Amb.	CE / Dry Amb.	W 4-Loop / Ice Cond.	W 2-Loop / Dry Amb.	W 3-Loop / Dry Sub.	W 3-Loop / Dry Amb.	W 4-Loop / Dry Sub.	W 4-Loop / Dry Amb.
1	ANO 1	ANO 2	Catawba 1	Ginna	Beaver Valley 1	Robinson 2	Millstone 3	Braidwood 1
2	Crystal River 3	Calvert Cliffs 1	Catawba 2	Kewaunee	Beaver Valley 2	Farley 1		Braidwood 2
4	Davis-Besse	Calvert Cliffs 2	D.C. Cook 1	Point Beach 1	North Anna 1	Farley 2		Byron 1
5	Oconee 1	Fort Calhoun	D.C. Cook 2	Point Beach 2	North Anna 2	Shearon Harris		Byron 2
6	Oconee 2	Millstone 2	McGuire 1	Prairie Island 1	Surry 1	Summer		Callaway
7	Oconee 3	Palisades	McGuire 2	Prairie Island 2	Surry 2	Turkey Point 3		Comanche Peak 1
8	TMI 1	Palo Verde 1	Sequoyah 1			Turkey Point 4		Comanche Peak 2
9		Palo Verde 2	Sequoyah 2					Diablo Canyon 1
10		Palo Verde 3	Watts Bar					Diablo Canyon 2
11		San Onofre 2						Indian Point 2
12		San Onofre 3						Indian Point 3
13		St. Lucie 1						Salem 1
14		St. Lucie 2						Salem 2
15		Waterford 3						Seabrook
16								South Texas 1
17								South Texas 2
18								Vogtle 1
19								Vogtle 2
20								Wolf Creek
	7	14	9	6	6	7	1	19
	7	14	9	19			20	

Total # of PWRs = 69

Notes

* Reviews are still need to be performed to determine grouping of these plants. After the reviews are completed, 2–3 additional MELCOR classes may be formed from the plants currently listed in Class 7.

Attachment 1

Mark I BWRs Internal Events Screening

Date Modified: 12/01/06

Plant Name	Total CDF	LLOCA	MLOCA	SLOCA	ISLOCA	ATWS	Scenario 1	Scenario 2	Scenario 3	Scenario 4	Scenario 5	Scenario 6
Plant 1	4.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 2	3.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 3	1.E-05	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 4	8.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 5	4.E-06	[redacted]	Note 1	[redacted]	[redacted]	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 6	4.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 7	2.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 8	5.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 9	1.E-05	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 10	3.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 11	4.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 12	5.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 13	1.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 14	1.E-05	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	Note 3	[redacted]	[redacted]
Plant 15	1.E-05	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 16	8.E-07	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

***Scenario Descriptions*:**

1. Reactor transients with unavailabilities of high-pressure injection systems (HPCI/RCIC) and RCS depressurization.
2. Station blackout with unavailability of high-pressure injection systems (HPCI/RCIC) and the failure of operators to recover emergency power within 30 minutes.
3. Station blackout with failure of operators to recover emergency power prior to battery depletion. This scenario could have sequence contributors with and without successful shedding of DC loads to extend the battery life.
4. Reactor transients with unavailabilities of RHR which leads to the unavailabilities of SPC/SDC/CSS, along with unavailabilities of containment venting or late injection. This scenario includes non-recoverable losses of service water/CCW.
5. Reactor transients with common-cause failure of the SRVs to open. This scenario is a plant-specific scenario to Plant 15 (i.e., derived from licensee PRA).
6. Reactor transients with common-cause failure of the transformer power supply inverters leads to the unavailabilities of all high- and low-pressure injection systems. This scenario is a plant-specific scenario for Plant 15 (i.e., derived from licensee PRA).

***Notes*:**

1. There is no MLOCA event tree for Plant 5.
2. The relatively high ATWS CDFs for Plants 5 and 8 are due to conservative modeling assumptions in these SPAR models. These modeling artifacts are currently being corrected by INL.
3. Plant 14 has dominant sequences with and without a stuck-open SRV. The CDF sum for the sequences involving a stuck-open SRV equal 4×10^{-8}. The CDF sum for the sequences involving a stuck-open SRV equal 3×10^{-8}.

Westinghouse 4-Loop, Large Dry PWRs Internal Events Screening

Date Modified: 12/01/06

Plant Name	Total CDF	LLOCA	MLOCA	ATWS	Scenario 1	Scenario 2	Scenario 3	Scenario 4	Scenario 5a	Scenario 5b	Scenario 6	Scenario 7
Plant 1	2.E-05	[redacted]	[redacted]	[redacted]	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 2	3.E-05	[redacted]	[redacted]	Note 1	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 3	5.E-06	[redacted]	[redacted]	[redacted]	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 4	5.E-05	[redacted]	[redacted]	[redacted]	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 5	4.E-05	[redacted]	[redacted]	[redacted]	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 6	9.E-06	[redacted]	[redacted]	[redacted]	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 7	9.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 8	8.E-06	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 9	5.E-06	[redacted]	[redacted]	[redacted]	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 10	1.E-05	[redacted]	[redacted]	Note 1	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 11	5.E-05	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 12	4.E-05	[redacted]	[redacted]	[redacted]	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Plant 13	4.E-05	[redacted]	[redacted]	[redacted]	Note 2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

***Scenario Descriptions*:**

1. ISLOCA from the RHR system.
2. Steam generator tube rupture (initiating event).
3. Reactor transients with unavailabilities of AFW and bleed and feed.
4. Station blackout with failure of turbine-driven AFW pump and the failure of operators to recover emergency power within 1 hour.
5. (a) Station blackout with failure of operators to recover emergency power prior to battery depletion.
   (b) Station blackout with RCP seal failure (LOCA) and failure to recover power prior to battery depletion time or 4 hours (which ever is less).
6. Loss of service water or CCW (non-recoverable or operators fail to recover) with failure of RCP seals (LOCA).
7. SLOCA with failure of RHR/HPR or RHR/LPR.

	Cont. Bypass	No Cont. Bypass
[redacted]	[redacted]	[redacted]
Red	CDF ≥ 1E-7	CDF ≥ 1E-6

***Notes*:**

1. The relatively high ATWS CDFs for Plants 2 and 10 are due to conservative modeling assumptions contained in these SPAR models. These modeling artifacts are currently being corrected by INL.
2. ISLOCA is only calculated for 3 of the 12 plants within this group. However, due to the future use of the same ISLOCA event tree for all PWR SPAR models and similar valve orientations, the ISLOCA CDFs for all plants within this group are expected to be in the range of the three completed plants.

Revised: December 7, 2006

STATUS
ACRS REVIEW OF HIGH PRIORITY STANDARD REVIEW PLAN SECTIONS

SRP SECTION	SRP Section Title	Mbr/Eng	Received	Status
2.3.1	Regional Climatology (See RG 1.76)	MC/MAJ	11/28/06	
2.3.3	Onsite Meteorological Measurements (See RG 1.23)	TSK/DCF	12/5/06	
2.4.6	Probable Maximum Tsunami Flooding	DAP/CGH		
2.5.2	Vibratory Ground Motion (See DG-1146)	DAP/HPN		
3.2.1	Seismic Classification	GEA/HPN	draft 10/5/06 formal 11/6/06	Don't Review
3.2.2	System Quality Group Classification	JSA/MB	draft 10/5/06 formal 11/6/06	Don't Review (verify with JSA)
3.12	ASME Code Class 1, 2, and 3 Piping Systems and Associated Supports Design [new] (See DG-1144)	JSA/CXS	draft 10/17/06	Don't Review
3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3	WJS/MB	draft 10/17/06 formal 11/6/06	Don't Review
4.2	Fuel System Design	JSA/RC		

[redacted]

5.4.8	Reactor Water Cleanup System (BWR)	JDS/MAJ	draft 10/23/06	TBD: Don't Review (tent)
6.2.5	Combustible Gas Control Containment	WJS/EAT	10/2/06	Letter 11/17/06
7.8	Diverse I&C Systems	GEA/EAT		
BTP 7-19	Guidance for the Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems	GEA/EAT		
9.1.1	New Fuel Storage	JSA/RC		
9.1.2	Spent Fuel Storage	TSK/HPN		
9.1.3	Spent Fuel Pool Cooling and Cleanup System	DAP/HPN	8/25/06	Don't Review Larkinsgram 11/6/06
9.5.1	Fire Protection Program (See RG 1.189)	JDS/MAJ	draft 10/25/06 formal 11/6/06	Revisit ~~Dec.~~ Feb
10.3.6	Steam and Feedwater System Materials	JSA/CXS	8/24/06	Don't Review Larkinsgram 11/6/06
11.2	Liquid Waste Management Systems	JDS//MB		
11.3	Gaseous Waste Management Systems	JDS/MB		
11.4	Solid Waste Management Systems	JDS/MB		
12.3 -12.4	Radiation Protection Design Features	DAP/CGH		
13.3	Emergency Planning	MC/MB	9/19/06	Review in Dec.
15.0	Accident Analysis - Introduction	SB/RC		
15.9	BWR Core Stability [new]	SB/RC		

17.4	Reliability Assurance Program [new]	GEA/EAT	10/31/06	Don't Review (tent.)
17.5	Quality Assurance	TSK/DCF	9/22/06	Don't Review Larkinsgram 11/6/06
19.0	Probabilistic Risk Assessment	GEA/EAT		

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
538th	10	1

Title **RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS**

List of Documents Attached	
See attached list	**10**

Instructions to Preparer	Lead Staff Person
1. Paginate Attachments 2. Punch holes 3. Place Copy in file box	SAM DURAISWAMY

SUBJECT	ANALYSIS	EDO LTR.	ACRS LTR.
Final Draft NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications (HPN/GEA)	11/29/06 (pp. 1-2)	11/27/06 (pp.3-4)	10/25/06 (pp. 5-9)
Browns Ferry Nuclear Plant, Unit 1 - Extended Power Uprate Application and Supplemental Application (RC/MVB)	12/07/06 (pp. 10-11)	12/01/06 (pp. 12-16)	11/06/06 (pp.17-18)
Report of the Safety Aspects of the License Renewal Application for the Palisades Nuclear Power Plant (MAJ/JDS)	12/08/06 (p. 19)	12/06/06 (p. 20)	11/27/06 (pp. 21-26)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

November 29, 2006

MEMORANDUM TO: George E. Apostolakis, Chairman
Reliability and Probabilistic Risk Assessment Subcommittee

FROM: H. P. Nourbakhsh, Senior Staff Engineer
ACRS

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON FINAL DRAFT NUREG-1824, "VERIFICATION AND VALIDATION OF SELECTED FIRE MODELS FOR NUCLEAR POWER PLANT APPLICATIONS"

Attached for your perusal is copies of the EDO's November 27, 2006 letter, responding to ACRS's October 25, 2006 letter concerning final draft NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." A copy of the ACRS's October 25, 2006 report is also attached.

Committee Report

In its letter, the Committee summarized its recommendations and comments on the final draft NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." Following are the Committee's recommendations:

1. The report provides a systematic evaluation of the predictive capability of five commonly used compartment fire models. It should be published.

2. The user's guide to be developed by the staff should include:

   a. Estimates of the ranges of normalized parameters to be expected in nuclear plant applications.
   b. Quantitative estimates of the uncertainties associated with each model's predictions, preferably in the form of probability distributions.

The Committee also noted that this commendable effort to validate models of compartment fires is an important first step in developing the fire models needed by the NRC to assess fire risks and licensee proposals. The Committee further noted the need for validated models of the effects of fires on equipment and cables as well as the need for the models of smoke transport within plants and the effects of deposited smoke on equipment and structures.

EDO Response

The EDO's response, dated November 27, 2006, touched on the Committee's letter of October 25, 2006, providing Committee's views on the final draft NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." The staff agrees

with both recommendations made by the Committee. The staff also agrees that NRC should continue to perform needed research activities in the area of fire modeling.
The EDO's response noted that the staff is discussing the development of an NPP fire modeling user's guide with EPRI and NIST, as a collaborative project. The staff will consider the points recommended by the Committee regarding "estimates of the ranges of normalized parameters to be expected in nuclear plant applications" and the "quantitative estimates of the uncertainties associated with each model's predictions, preferably in the form of probability distributions." The EDO's response further noted that toward that end, the staff looks forward to interacting with the ACRS again throughout the development of the fire modeling user's guide.

Analysis

The staff has agreed to ACRS recommendations. The Committee will be afforded opportunities to discuss the development of fire modeling user's guide as the work progresses.

Attachments: As Stated

cc w/o attach (via E-mail):
ACRS Members
J. Larkins
F. Gillespie
J. Flack
M. Snodderly
C. Santos
S. Duaiswamy
ACRS Technical Staff

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 27, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: RESPONSE TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) LETTER, DATED OCTOBER 25, 2006, CONCERNING DRAFT FINAL NUREG-1824, "VERIFICATION AND VALIDATION OF SELECTED FIRE MODELS FOR NUCLEAR POWER PLANT APPLICATIONS"

Dear Dr. Wallis:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter, dated October 25, 2006, concerning NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." We agree with the ACRS that NRC should publish this collaboratively prepared NUREG-series report, because it provides applicable information for use by the NRC staff and the nuclear industry. We also agree that NRC should continue to perform needed research activities in the area of fire modeling.

As noted in your letter, the NRC's Office of Nuclear Regulatory Research collaborated with the Electric Power Research Institute (EPRI) and the National Institute of Standards and Technology (NIST) in conducting this fire model verification and validation (V&V). This work was the first of its kind in performing a systematic, detailed fire model V&V in accordance with the American Society for Testing and Materials International Standard E-1355, "Evaluating the Predictive Capability of Deterministic Fire Models." In addition to supporting the current fleet of nuclear power plants (NPP) in the use of fire models for fire hazards analysis or license exemption requests, the report directly supports the NRC's new risk-informed, performance-based fire protection rule set forth in Title 10, Section 50.48(c)), of the *Code of Federal Regulations* (10 CFR 50.48(c)). In particular, that rule endorses the "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants," which the National Fire Protection Association (NFPA) has promulgated as NFPA Standard 805.

The staff acknowledges and agrees with the ACRS recommendation that the next step in fire modeling should be the development of an NPP fire modeling user's guide. The staff is discussing this guide with EPRI and NIST, as a collaborative project. We also will consider the points recommended in your letter regarding the "estimates of the ranges of normalized parameters to be expected in nuclear plant applications" and the "quantitative estimates of the uncertainties associated with each model's predictions, preferably in the form of probability distributions." Toward that end, the staff looks forward to interacting with the ACRS again throughout the development of the fire modeling user's guide.

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In closing, we value the review and comments that the ACRS provided regarding this report. We also appreciate your commendation of the organizations and individuals involved in preparing the NUREG-series report.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

October 25, 2006

Mr. Luis Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

SUBJECT: DRAFT FINAL NUREG-1824, "VERIFICATION AND VALIDATION OF SELECTED FIRE MODELS FOR NUCLEAR POWER PLANT APPLICATIONS"

Dear Mr. Reyes:

During the 536th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2006, we met with representatives of the NRC staff, Electric Power Research Institute (EPRI), and the National Institute of Standards and Technology (NIST) to discuss the draft final NUREG-1824 (EPRI 1011999), "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." Our Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) also reviewed this matter during its meeting on September 21, 2006. During our review, we had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATIONS

1. The report provides a systematic evaluation of the predictive capability of five commonly used compartment fire models. It should be published.

2. The user's guide to be developed by the staff should include:

    a. Estimates of the ranges of normalized parameters to be expected in nuclear plant applications.
    b. Quantitative estimates of the uncertainties associated with each model's predictions, preferably in the form of probability distributions.

BACKGROUND

Fire models are used in a number of safety evaluations, including fire risk analysis; demonstrating compliance with, and exemptions to, the regulatory requirements for fire protection in 10 CFR Part 50, Appendix R; the significance determination process of the Reactor Oversight Process; and establishing the risk-informed, performance-based voluntary

5

fire protection licensing basis under 10 CFR 50.48(c) and the referenced 2001 Edition of the National Fire Protection Association (NFPA) Standard, NFPA 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Stations." NFPA 805 requires that "only fire models that are acceptable to the authority having jurisdiction shall be used in fire modeling calculations." NFPA 805 further requires that the fire models be verified and validated, and be applied only within their domains of validity.

The NRC Office of Nuclear Regulatory Research (RES) and EPRI sponsored a collaborative project for the verification and validation of selected fire models that are commonly used in the nuclear industry. NIST participated in this work. Report NUREG-1824 (EPRI 1011999) is the result of this collaborative project.

The selected models are:

- Fire Dynamics Tools (FDTs) developed by the NRC
- Fire-Induced Vulnerability Evaluation, Revision 1 (FIVE-Rev1) developed by EPRI
- Consolidated Model of Fire Growth and Smoke Transport (CFAST) developed by NIST
- MAGIC developed by Electricité de France (EdF)
- Fire Dynamics Simulator (FDS) developed by NIST

The verification and validation study was based on the methodology described in the American Society for Testing and Materials (ASTM) International Standard E 1355 - 05a "Standard Guide for Evaluating the Predictive Capability of Deterministic Fire Models."

A draft version of NUREG-1824 was issued for public comment on January 31, 2006. The comment period closed on March 31, 2006. The project team responded to all of the public comments.

DISCUSSION OF THE NUREG REPORT

Ever since the Browns Ferry fire in 1975 and the publication of several PRAs that demonstrated the risk significance of fires, there has been a great deal of interest in modeling the effects of fire on nuclear power plants. A number of deterministic models have been proposed focusing primarily on compartment fires. These are based on varying assumptions and calculational methods ranging from simple hand calculations (FIVE-Rev1 and FDTs) to two-zone models (CFAST and MAGIC) to sophisticated detailed models (FDS). This study is the first systematic evaluation of the ability of fire models to predict experimental results and will be very useful to both the NRC and the industry.

The project team identified 13 parameters that are likely to be required in safety assessments involving fires. These parameters were selected by reviewing potentially risk-significant scenarios from a variety of sources and are limited to those that describe the environment created by a fire in a compartment, e.g., the height and temperature of the hot gas layer, the flame height, the smoke concentration, and the radiant heat flux. This set of parameters does not characterize other important fire phenomena that are out of the scope of the present work, such as fire propagation in cable trays.

(6)

The ability of the selected models to estimate numerical values for the chosen parameters was evaluated by comparing their results with experimental measurements. The measured heat release rates from the fires were used as input to the analyses. Twenty-six experiments were selected from five test series that were judged to be relevant to nuclear plant applications and for which sufficient information was available to allow quantitative evaluations. The experiments were performed using pool fires with a variety of hydrocarbon fuels and a wide range of heat release rates.

The model predictions for each experiment were compared with the experimental results. There are uncertainties associated with these comparisons because of uncertainty in model input (primarily the heat release rate) and uncertainty in the measurements themselves. The experimental *measurement uncertainty* and the experimental *model input uncertainty* are used to develop a range of possible values of the scenario parameter of interest. The accuracy of the model predictions is qualitatively characterized by a simple color code.

DISCUSSION OF THE USER'S GUIDE

The staff plans to develop a user's guide to complement NUREG-1824. A user will have to determine whether the results of the verification and validation study are applicable to the situation to be analyzed. This is done using "normalized parameters" (i.e., governing non-dimensional groups, not to be confused with the 13 scenario parameters discussed above) that allow users to compare results from scenarios of different scales by normalizing physical characteristics of the scenario. These normalized parameters are traditionally used in fire modeling applications and are included in the NUREG report. The user's guide should provide estimates of the ranges of normalized parameters to be expected in nuclear plant applications. These estimates would allow a determination of whether risk-significant fires fall within or outside the parameter ranges covered by the verification and validation process.

The user's guide should also provide probability distributions for the model predictions due to the intrinsic model uncertainty, i.e., the uncertainty associated with the model's physical and mathematical assumptions. These distributions should not include the uncertainties in the heat release rate since the latter will be an input specified by the user. The color designations provide no quantitative estimate of the intrinsic uncertainty. This uncertainty is an important input in risk-informed applications. Even in non-risk-informed applications, a quantitative assessment of the tendency of a model to over- or under-predict would be valuable. The staff told us that such quantitative estimates will be provided in the user's guide. We look forward to reviewing this document.

CONCLUDING REMARKS

We commend the RES staff and EPRI for undertaking this project and providing the basis for the evaluation of fire models. The NUREG report and the user's guide will significantly improve the technical basis supporting the fire safety evaluations.

(7)

This commendable effort to validate models of compartment fires is an important first step in developing the fire models needed by the NRC to assess fire risks and licensee proposals. Validated models of the effects of fires on equipment and cables are needed. Also needed are models of smoke transport within plants and the effects of deposited smoke on equipment and structures. We look forward to interacting with the staff as this research progresses.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References

1. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Vol 1: Main Report*, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

2. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Vol 2: Experimental Uncertainty,* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

3. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Vol 3: Fire Dynamics Tools (FDT^s),* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

4. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Vol 4: Fire-Induced Vulnerability Evaluation (FIVE-Rev1),* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

5. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Application EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Vol 5: Consolidated Fire Growth and Smoke Transport (CFAST),* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

(8)

6. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Application EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Vol 6: MAGIC,* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

7. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Vol 7: Fire Dynamics Simulator (FDS),* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

8. NFPA 805,"Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Stations," 2001 Edition, National Fire Protection Association, Quincy, MA.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

December 7, 2006

MEMORANDUM TO: Mario V. Bonaca, Chairman
Power Uprate Subcommittee

FROM: R. Caruso, Senior Staff Engineer
ACRS

SUBJECT: ANALYSIS OF NRR RESPONSE TO ACRS MEMORANDUM CONCERNING THE BROWNS FERRY NUCLEAR STATION 105% POWER UPRATE

Attached for your information is a copy of NRR's December 1, 2006 response to the ACRS's memorandum of November 7, 2006, concerning the Committee's request to review the 105% power uprate for Browns Ferry Unit 1 (BFN1). A copy of the Committee's letter is also attached.

Committee Memorandum

In its memorandum, the Committee informed the EDO that the ACRS had decided to review the 105% power uprate for BFN1.

NRR Response

The staff response provided the ACRS with the draft Safety Evaluation for the 105% power uprate application. In addition, the staff noted that although the licensee had performed bounding analyses at the 120% OLTP level to support this uprate, those " ... bounding analyses (120-percent level) would not necessarily imply staff approval of the analyses for operation at the 120-percent level." The staff further proposed that since (1) the analyses were performed at 120 percent, and (2) the staff's review methodology is the same regardless of the power level, and (3) the three Browns Ferry plants are essentially the same in design and operation, the ACRS could perform its review of the 105% safety evaluation for Unit 1, and limit consideration of the 120% uprates for Units 1,2,3, and 3 to only those issues that are specific to 120% operation. The staff is planning to support this safety evaluation before the Power Uprate Subcommittee in January, and the full Committee in February 2007.

Analysis

The staff's response to provide the ACRS with the SER is acceptable. I will provide the members with a CD that includes the safety evaluation and all of the supporting documentation, including the internal staff SER inputs. I will provide a separate status report with my analysis of the SE by December 19, 2006.

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The staff proposal to limit future ACRS consideration of the BF uprate request to "... those additional 120-percent topics applicable for Units 1,2, and 3", is unacceptable.

The reactor safety analyses for BFN1 were performed by the BFN1 fuel vendor, General Electric. BFN2 and BFN3 are fueled with Areva fuel, and Areva is therefore responsible for a substantial fraction of the licensing-basis analyses supporting operation of those plants at EPU conditions. Therefore, the BFN1 analyses and SE are not applicable to BFN2/3.

This proposal also contradicts the review plan that the staff described in its letter to TVA on October 17, 2006. In that document, the staff stated that it would review the application for the 105% uprate, and document those results in a safety evaluation, but this safety evaluation would not contain any conclusions regarding the acceptability of the analyses in support of operation at EPU conditions.

> "The NRC staff will subsequently review the information supplied in support of your EPU application in its entirety [emphasis added] to determine if there is reasonable assurance that operation at EPU is consistent with the Commission's regulations. This review will be documented in a separate, stand-alone safety evaluation that will specifically address each topic delineated in the template safety evaluation provided in the Review Standard. The amendment package, including the safety evaluation, will be reviewed in accordance with standard practice, which includes review by the Advisory Committee for[sic] Reactor Safeguards."

This letter was sent to TVA in response to the TVA proposal on September 22, 2006, to do essentially the type of review that the staff now proposes for the ACRS. The staff has even taken the TVA one step further, by proposing that it apply to all three BF units, not just Unit 1.

I would note that he wording of the October 17 letter is curious. It states that the staff will determine whether the application "is consistent with the Commission's regulations". In order to issue a license amendment, the staff must make a finding that:

> "The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;"

These words are the standard boilerplate in every license amendment issued by NRR. I do not understand how the staff goes from establishing that the application "is consistent with" to "in conformity with" the regulations.

Finally, the staff may say that BFN1 is "essentially the same in design and operation" as BFN2/3, but it is really not. It has been substantially rebuilt, and now includes many new systems and components that are different from those in BFN2/3. It also includes a large amount of abandoned-in-place wiring that was completely replaced (~850,000 feet), as the Committee noted during its visit in 2005.

In any case, although acquiescence with the staff's proposal might save staff and ACRS resources, I do not believe that it is prudent. The Committee should decline this offer, and plan to review the 120% uprates for all three Browns Ferry units.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 1, 2006

DEC - 5 2006

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

FROM: J.E. Dyer, Director
Office of Nuclear Reactor Regulation

SUBJECT: TRANSMITTAL OF THE BROWNS FERRY 105-PERCENT POWER LEVEL UPRATE SAFETY EVALUATION FOR UNIT 1

On November 7, 2006, the Advisory Committee on Reactor Safeguards (ACRS), informed the Office of Nuclear Reactor Regulation (NRR) that in view of Tennessee Valley Authority's (TVA's) use of bounding arguments to demonstrate that the safety analyses performed at 120-percent power level can be used to support operation at 105 percent, the ACRS has decided to review the 105-percent Safety Evaluation (SE). Accordingly, I am providing a copy of the draft SE (ADAMS Accession No. ML063350404), for ACRS sub- and full-committee review in January and February 2007, respectively. This will support TVA's current restart plan.

Although bounding analyses (120-percent level) would not necessarily imply staff approval of the analyses for operation at the 120-percent level, and thus, ACRS waiver of review of 105 percent may be justified, the staff recognizes the benefits of ACRS review of the 105-percent SE. Since TVA's analyses were performed at 120 percent and the staff's review methodology is essentially the same regardless of the power level, ACRS review of the 120-percent SE in addition to 105-percent SE, may only be necessary for those issues (e.g., steam dryer, safety limits, etc.) that are different from the 105-percent SE. Also, TVA has indicated that Units 1, 2 and 3, are essentially the same in design and operation. Therefore, we propose ACRS review of the Unit 1 SE for 105 percent, and limited review of those additional 120-percent topics applicable for Units 1, 2, and 3. The staff requests ACRS consideration of this approach, which would avoid review duplication and result in significant savings of ACRS and NRR staff resources.

It should be noted that the attached draft SE may contain proprietary information and, therefore, should not be released to the public. The NRR staff will provide the draft SE to the licensee to ensure that any proprietary information is appropriately identified and controlled. A publicly available version will be released after completion of the proprietary review.

I thank you for your expeditious review and consideration of this approach.

Docket No. 50-259

Enclosures:
1. Draft SE
2. Diskette w/Background Information

cc: w/o encl: Bill Kane
L. Reyes

CONTACT: E. Brown, DORL/LPL2-2
301-415-2315
EAB1@nrc.gov

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October 17, 2006

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 — REVIEW OF PROPOSED AMENDMENT FOR FIVE PERCENT INCREASE IN THERMAL POWER (TAC NO. MD3048) (TS-431)

Dear Mr. Singer:

The purpose of this letter is to inform you of the U.S. Nuclear Regulatory Commission (NRC) staff's review methodology for your September 22, 2006, license amendment request. This methodology was discussed with your staff on October 10, 2006. On September 22, 2006, the Tennessee Valley Authority (the licensee) submitted a request to supplement the Browns Ferry Nuclear Plant (BFN) Unit 1 June 28, 2004, request. The June 28, 2004, amendment request involves a change in licensed thermal power from 3293 megawatt thermal (MWt) to 3952 MWt, an approximate 20 percent increase in thermal power (commonly referred to as an extended power uprate (EPU)). The September 22, 2006, supplement requested approval of an increase in licensed thermal power from 3293 MWt to 3458 MWt with an attendant 30-pounds per square inch increase in reactor pressure. This represents an approximate 5-percent increase above the original licensed thermal power (OLTP) of 3293 MWt, and is commonly referred to as a stretch power uprate. The NRC staff has reviewed the information provided and determined that sufficient information has been provided to begin the technical review of your application.

In the submittal, TVA states that the analyses and evaluations previously performed for EPU operation at 120 percent OLTP in the PUSAR (Power Uprate Safety Analysis Report) support and/or bound operation at lesser power levels, subject to the operating restrictions and limitations that would be applicable. In other cases, the staff requested and you have agreed to provide cycle-specific analyses (105% power level) e.g, safety limit minimum critical power ratio. In its review of your stretch power uprate request, the NRC staff will use the information supplied in support of your EPU application to determine if there is reasonable assurance that operation at the stretch power uprate conditions is consistent with the Commission's regulations. The results of this review will be documented in a safety evaluation. This safety evaluation will not contain any conclusions regarding the acceptability of your analysis in support of operation at EPU conditions.

The NRC staff will subsequently review the information supplied in support of your EPU application in its entirety to determine if there is reasonable assurance that operation at EPU conditions is consistent with the Commission's regulations. This review will be documented in a separate, stand-alone safety evaluation that will specifically address each topic delineated in

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the template safety evaluation provided in the Review Standard. The amendment package, including the safety evaluation, will be reviewed in accordance with standard practice, which includes review by the Advisory Committee for Reactor Safeguards.

If you have any questions, please contact the BFN Unit 1 Project Manager, Ms. Margaret Chernoff, at (301) 415-4041.

Sincerely,

/RA/

L. Raghavan, Chief
Project Directorate II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

cc: See next page

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the template safety evaluation provided in the Review Standard. The amendment package, including the safety evaluation, will be reviewed in accordance with standard practice, which includes review by the Advisory Committee for Reactor Safeguards.

If you have any questions, please contact the BFN Unit 1 Project Manager, Ms. Margaret Chernoff, at (301) 415-4041.

Sincerely,

/RA/

L. Raghavan, Chief
Project Directorate II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

cc: See next page

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Mr. Karl W. Singer
Tennessee Valley Authority

BROWNS FERRY NUCLEAR PLANT

cc:

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(14)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

November 6, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 - EXTENDED POWER UPRATE APPLICATION AND SUPPLEMENTAL APPLICATION

On June 28, 2004, Tennessee Valley Authority (TVA) submitted to the staff an amendment request to raise the thermal power of Browns Ferry Nuclear Plant (BFN) Unit 1 from 3293 MWt to 3952 MWt, an approximate 20% power increase in original licensed thermal power (OLTP) (Reference 1). This is commonly referred to as an extended power uprate (EPU)). Because of concerns with steam dryers operation at the EPU level, TVA will need to gather data and perform analyses to support staff completion of its SER. This will delay completion of the SER until data and analyses are provided to the staff. The ACRS plans to review this EPU as soon as the related final SER becomes available.

On September 22, 2006, TVA submitted an amendment supplement (Reference 2) requesting approval of an increase in licensed thermal power of approximately 5% above the OLTP (referred to as a stretch power uprate). TVA stated that it will use the analyses performed at the 120% OLTP to license operation at 105% OLTP whenever the analyses performed at 120% OLTP bound those performed at 105% OLTP. In its amendment supplement TVA stated that after review and approval of the 105% OLTP power uprate, the transition to 120% OLTP will..." only be contingent upon NRC review and acceptance of the steam dryer stress report. All other safety evaluations that support operation at 105% OLTP would remain valid for operation at 120% OLTP".

Normally, the ACRS does not review power uprates less than about 105% OLTP (Reference 3). But in the case of BFN Unit 1, the licensee will use bounding arguments to demonstrate that safety analyses performed at 120% OLTP are valid to support operation at 105% OLTP. In view of the intent of the licensee to take this approach, the ACRS has decided to review the SER for the 105% power uprate for BFN1.

(11)

References:

1. Letter dated June 28, 2004 from T. Abney to Document Control Desk, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Proposed Technical Specifications (TS) Change TS-431 - Request for License Amendment - Extended Power Uprate (EPU) Operation"

2. Letter dated September 22, 2006 from W. Crouch to Document Control Desk, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Technical Specifications (TS) Change TS-431, Supplement 1 - Extended Power Uprate (EPU) (TAC No. MC3812)"

3. Memorandum dated October 9, 2003, from John T. Larkins to James E. Dyer, "Kewaunee Nuclear Power Plant - Advisory Committee on Reactor Safety Review of Stretch Power Uprate Amendment (TAC No. MB9031)"

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
B. Sosa, OEDO
J. Lamb, OEDO
J. Dyer, NRR
C. Haney, NRR
C. Holden, NRR
L. Raghavan, NRR
M. Chernoff, NRR
E. Brown, NRR
M. Zobler, OGC

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

December 8, 2006

MEMORANDUM TO: John D. Sieber, Chairman
License Renewal Subcommittee

FROM: Michael R. Junge, Senior Staff Engineer
Advisory Committee on Reactor Safeguards Staff

SUBJECT: ANALYSIS OF EDO RESPONSE TO THE ACRS LETTER, DATED DECEMBER 6, 2006, CONCERNING THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE PALISADES NUCLEAR PLANT

Attachment 1 contains a copy of the Executive Director for Operations (EDO) December 6, 2006 response to the Advisory Committee on Reactor Safeguards (ACRS) November 17, 2006 letter regarding the Safety Aspects of the License Renewal Application for the Palidades Nuclear Plant (PNP). Attachment 2 contains a copy of the Committee letter.

Recommendation 1

The NMC application for renewal of the operating license for PNP should be approved.

EDO Response

The staff agrees with the Committee's recommendation to renew the operating license for PNP.

Analysis

The EDO agrees with the ACRS recommendation.

cc: ACRS Members
J. Larkins
M. Snodderly
S. Duraiswamy
C. Santos

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 6, 2006

Dr. Graham B. Wallis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RESPONSE TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE PALISADES NUCLEAR PLANT

Dear Dr. Wallis:

During the 537th meeting of the Advisory Committee on Reactor Safeguards (ACRS or the Committee) held on November 1, 2006, the ACRS completed its review of the license renewal application (LRA) for the Palisades Nuclear Plant (PNP) and the associated final safety evaluation report (SER) prepared by the U.S. Nuclear Regulatory Commission (NRC) staff. In its final report, the Committee recommends the renewal of the operating license for PNP, with the conclusions and recommendations discussed in your letter dated November 17, 2006. The staff appreciates the Committee's expeditious, objective, and in-depth review of the PNP application and the staff's final SER. The staff agrees with the Committee's conclusions:

1. The programs established and committed to by the applicant to manage age-related degradation provide reasonable assurance that PNP can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. The three Time-Limited Aging Analyses described in the Committee's letter must be addressed in accordance with NRC regulations during the period of extended operation.

3. Nuclear Management Company's application for renewal of the operating license for PNP should be approved.

The staff recognizes the ACRS's commitment to safety and appreciates the Committee's continued support of the license renewal process.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner McGaffigan
Commissioner Merrifield
Commissioner Jaczko
Commissioner Lyons
SECY

(20)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 17, 2006

The Honorable Dale E. Klein
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE PALISADES NUCLEAR POWER PLANT

Dear Chairman Klein:

During the 537th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2006, we completed our review of the license renewal application for the Palisades Nuclear Plant (PNP) and the final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on July 11, 2006. During our review, we had the benefit of discussions with representatives of the NRC staff and the applicant, Nuclear Management Company, LLC (NMC). In addition, we had the benefit of input from the public. We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

The programs established and committed to by the applicant to manage age-related degradation provide reasonable assurance that PNP can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

The NMC application for renewal of the operating license for PNP should be approved. Continued operation during the entire period of extended operation is contingent on the resolution of the issues associated with three Time-Limited Aging Analyses (TLAAs) related to reactor pressure vessel (RPV) integrity.

BACKGROUND AND DISCUSSION

PNP is a Combustion Engineering 2-loop pressurized water nuclear plant with a large, dry, ambient-pressure containment. PNP is located five miles south of South Haven, Michigan, on the eastern shore of Lake Michigan. The current power rating of the PNP

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is 2566 MWt, for a gross electrical output of 767 MWe. PNP was originally licensed to operate on February 21, 1971. NMC requested renewal of the PNP operating license for 20 years beyond the current license term, which expires on February 20, 2011.

In the final SER, the staff documented its review of the license renewal application and other information submitted by NMC and obtained during the audit and inspection conducted at the plant site. The staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive long-lived components; the adequacy of the applicant's Aging Management Programs (AMPs); and the identification and assessment of TLAAs requiring review.

The NMC application is largely consistent with NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," issued in July 2001. All deviations from the GALL Report are documented in the application. The applicant identified the SSCs that fall within the scope of license renewal and performed a comprehensive aging management review for these SSCs. Based on the results of this review, the applicant will implement 24 AMPs for license renewal including existing, enhanced, and new programs. In the final SER, the staff concluded that the applicant has appropriately identified the SSCs within the scope of license renewal and that the AMPs described by the applicant are appropriate and sufficient to manage aging of long-lived passive components that are within the scope of license renewal. We concur with this conclusion.

The staff conducted an inspection and an audit. The inspection verified that the scoping and screening methodologies are consistent with the regulations and are adequately reflected in the application. The audit verified the appropriateness of the AMPs and the aging management reviews. Based on the inspection and audit, the staff concluded that these programs are consistent with the descriptions contained in the NMC license renewal application. The staff also concluded that the existing programs, to be credited as AMPs for license renewal, are generally functioning well and that an implementation plan has been established in the applicant's commitment tracking system to ensure timely completion of the license renewal commitments.

During our meetings with the staff and the applicant, we discussed the adequacy of programs proposed by NMC to manage aging of certain components that are projected to exceed acceptance limits during the period of extended operation.

The applicant identified the systems and components requiring TLAAs and reevaluated them for 20 additional years of operation. As required by 10 CFR Part 54, the applicant must identify any exemptions granted under 10 CFR 50.12 which rely on a TLAA and determine if that exemption should be continued for an additional 20 years of operation. No such exemption currently exists in the PNP licensing basis. The applicant reexamined 23 TLAAs. All of these TLAAs are valid, without restriction, for 20 more years of operation, except for three TLAAs associated with reactor vessel neutron embrittlement, namely: reactor vessel upper shelf energy, reactor vessel pressurized thermal shock, and reactor vessel pressure-temperature curves. In each of these cases, PNP will exceed the acceptance limits prior to the end of the extended period of operation.

To analyze the reactor vessel neutron fluence for purposes of RPV integrity evaluations, the applicant uses the methodology described in WCAP-15353, which is consistent with Regulatory Guide 1.190.

The applicant began using low neutron leakage cores in 1988 to reduce the neutron embrittlement of the reactor vessel to extend the time before exceeding the acceptance limits. However, the applicant predicts that the following acceptance limits will be exceeded:

- Upper Shelf Energy limit – exceed in 2021.
- Reactor Vessel Pressurized Thermal Shock (PTS) screening criterion – exceed in 2014.
- Pressure-Temperature limit curves – expire in 2014.

The staff's confirmatory calculations show reasonable agreement with the applicant's findings.

Upper Shelf Energy Limit. The applicant predicts this criterion will be exceeded in 2021. Appendix G of 10 CFR 50 requires RPV beltline materials to have Charpy upper shelf energy values no less than 50 ft-lb in the transverse direction in the base metal and along a weld for weld material. However, in accordance with Appendix G, Charpy upper shelf energy values below 50 ft-lb may be acceptable if it is demonstrated that lower Charpy upper shelf energy values will provide margins of safety against fracture (ductile tearing) equivalent to those required by ASME Code, Section XI, Appendix G. Regulatory Guide 1.99 describes two acceptable methods for determining the upper shelf energy values for RPV beltline materials.

Because the reactor vessel upper shelf energy limit will be exceeded prior to the end of the extended period of operation, the applicant must provide an analysis in accordance with 10 CFR Part 50, Appendix G at least three years prior to exceeding the upper shelf energy limit.

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PTS Screening Criterion. The applicant predicts the criterion for axial welds and plates will be exceeded in 2014. 10 CFR 50.61 provides the fracture toughness requirements for protecting reactor vessels from the effects of PTS events. The end of life reference temperature (RT_{PTS}) value is the sum of a reference value for an unirradiated material, a shift in the reference value caused by exposure to high-energy neutron irradiation, and an additional margin to account for uncertainties.

If an applicant determines that the RPV will not meet the PTS screening criterion through the end of the facility's current license term, several actions must be taken. 10 CFR 50.61(b)(3), requires that an applicant implement a reasonably practicable flux reduction program in an effort to avoid exceeding the PTS screening criterion. If no reasonably practicable flux reduction program will meet this objective (as is true in the case of PNP) the applicant has several options. The applicant may submit a safety analysis in accordance with 10 CFR 50.61(b)(4) to demonstrate that the RPV can be operated beyond the 10 CFR 50.61 screening criterion. This safety analysis may include plant modifications. Such an analysis must be submitted three years prior to the time the RPV is projected to exceed the PTS screening criterion. In accordance with 10 CFR 50.61(b)(7), the applicant could propose to anneal the RPV in order to improve its material properties and permit continued operation. In accordance with 10 CFR 50.66, the applicant's thermal annealing plan would have to be submitted three years prior to when the facility's RPV is projected to exceed the PTS screening criterion.

Pressure-Temperature Limit Curves. Pressure-temperature limit curves are contained in the PNP technical specifications and are assessed against the limits in 10 CFR 50.60, Appendix G to 10 CFR 50, and Appendix G to Section XI of the ASME Code. The current pressure-temperature limits approved by the staff are valid beyond the current license term, but not through the extended period of operation. Based on the neutron fluence expected to be accumulated, the pressure-temperature limit curves will expire in 2014. Prior to entering the period of extended operation, the applicant must submit an amendment requesting a technical specification change and approval of new limits covering the period of extended operation beyond 2014.

The staff has concluded that the applicant has provided an adequate list of TLAAs. Further, the staff has concluded that the applicant has met the license renewal rule by demonstrating that the TLAAs have been projected to the end of the period of extended operation. In those cases where the current TLAAs do not cover the entire period of extended operation, the applicant must provide additional information in a timely manner and submit a license amendment for a technical specification change to extend these three TLAAs to cover the entire period of extended operation. We concur with the staff that the applicant has properly identified the applicable TLAAs, reviewed the

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associated analyses and licensing bases, and identified those instances where additional measures are needed to modify the TLAAs to cover the entire period of extended operation. We concur with the staff's conclusions and the resulting license conditions and commitments.

During our Plant License Renewal Subcommittee meeting on July 11, 2006, members of the Public provided comments and raised several questions. These comments and questions were recorded and are contained in the transcript of that meeting. The reference to the transcript that contains these comments and questions was provided to the Executive Director for Operations. Subsequently, the staff has responded to these questions and comments.

We agree with the staff that there are no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) that preclude renewal of the operating license for PNP. The programs established and committed to by NMC provide reasonable assurance that PNP can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. Continued operation during the entire period of extended operation is contingent on the resolution of the issues associated with three TLAAs related to RPV integrity. The NMC application for renewal of the operating license for PNP should be approved.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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References:

1. Safety Evaluation Report Related to the License Renewal of the Palisades Nuclear Power Plant, September 2006.
2. Palisades Nuclear Power Plant - Application for Renewed Operating Licenses, March 22, 2005
3. Safety Evaluation Report with Open Items Related to the License Renewal of the Palisades Nuclear Power Plant, June 2006
4. Audit and Review Report for Plant Aging Management Reviews and Programs (AMPs) (AMRs) - Palisades Nuclear Power Plant, October 20, 2005
5. Palisades Nuclear Power Plant, Inspection Report 05000255/2005009, December 28, 2005
6. Memorandum dated September 13, 2006, from John T. Larkins, Executive Director, ACRS, to Luis A. Reyes, Executive Director for Operations, Subject: Questions Raised by Members of the Public During the ACRS Subcommittee Meeting on Palisades Nuclear Plant License Renewal Application
7. Regulatory Guide 1.99 Revision 2, Radiation Embrittlement of Reactor Vessel Materials, May 1988
8. Regulatory Guide 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, March 2001
9. Palisades Reactor Pressure Vessel Fluence Evaluation, WCAP-15353, January 2000

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Proposed Revision to Section 13.3, "Emergency Planning" (EP) of the Standard Review Plan (SRP) & Combined License Application Guidance (DG-1145)

Presented By

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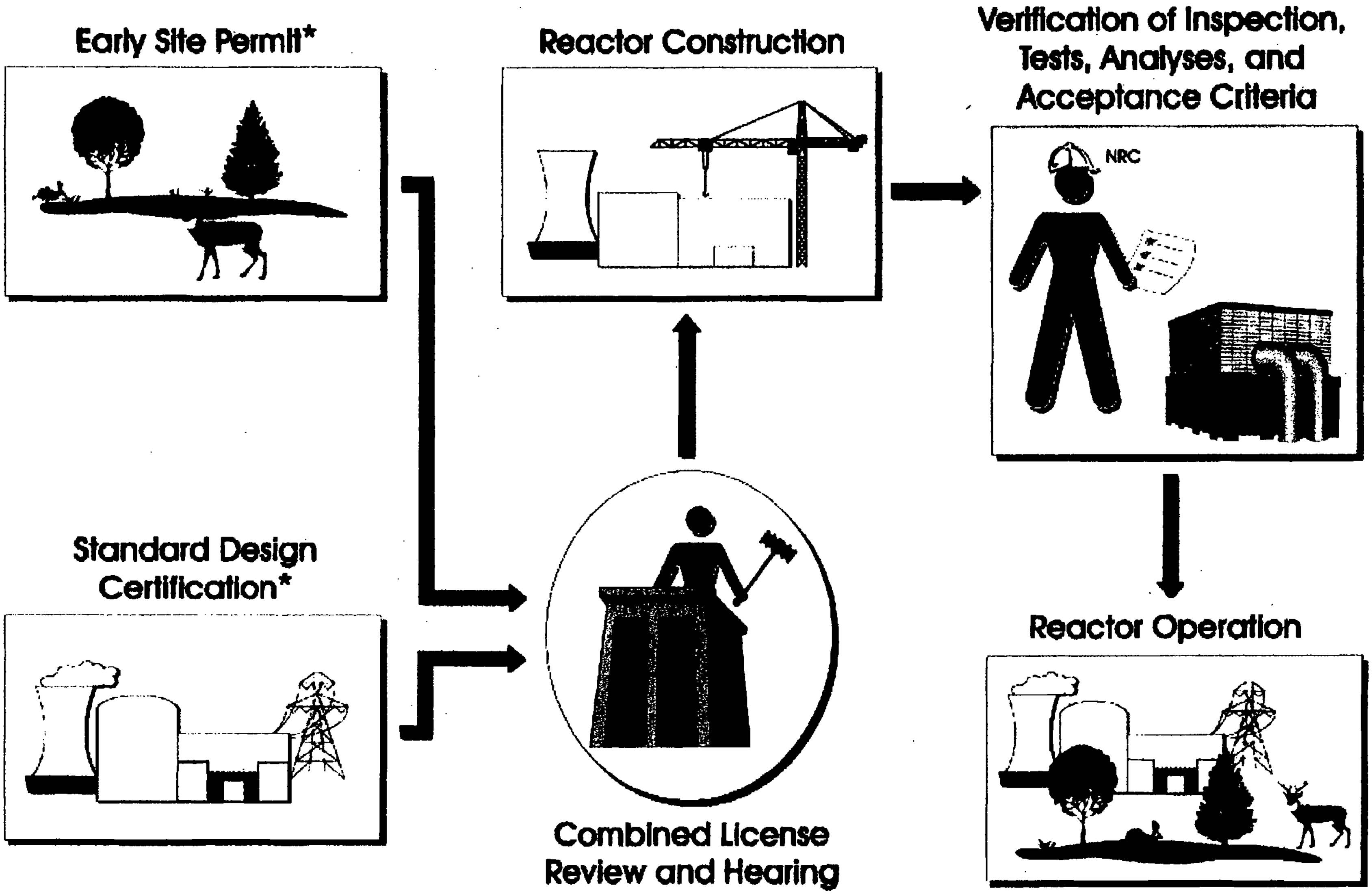
Section 13.3, "EP" of the SRP and DG-1145

- Standard Review Plan (SRP) (NUREG-0800) Section 13.3, "Emergency Planning"

- COL Application Guidance (DG-1145) Section 13.3, "Emergency Planning"

Combined Licenses, Early Site Permits, and Standard Design Certifications



* or equivalent process

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process

- Emergency Planning continues to be a part of the licensing process. (10 CFR 50.33, 50.34, 50.47, 50.54, and Appendix E, and 10 CFR part 52)
- President's decision of December 7, 1979 reemphasizes the NRC's continuing statutory responsibility for the radiological health and safety of the public.

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- Emergency Preparedness "Reasonable Assurance" finding needed prior to issuing License - 10 CFR 50.47(a)
- **NRC** makes this finding based on:
  - A review of FEMA (**DHS**) findings and determinations concerning **offsite** plans
  - and **NRC** findings and determinations concerning **onsite** plans.
  - NRC/FEMA Memorandum of Understanding establishes working relationship - 44 CFR 353, Appendix A

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- 16 Planning Standards of 10 CFR 50.47(b)
- Requirements of 10 CFR 50, Appendix E
- Regulatory Guide 1.101
    - NUREG-0654, FEMA-REP-1, Rev. 1 Acceptance Criteria
    - NEI 99-01, Rev. 4, EALs

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- Emergency Preparedness (EP)
  - A "Licensing Condition" - 10 CFR 50.54(q)
  - Deficiency 120 day clock - 10 CFR 50.54(s)
  - Reality presumption - 10 CFR 50.47(c)
  - Supported by two sets of plans:
    - "Onsite" emergency plan (Facility plan)
    - "Offsite" emergency plan (State & local plans)

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- **10 CFR Part 50**
  - **2-Step Process:**
    - Construction Permit
    - Operating License

An EP Perspective on New Reactor Licensing

Regulatory Process *(cont'd.)*

- **10 CFR Part 52 – Alternative licensing process**
  - **Established in 1989**
    - Improve Regulatory Efficiency
    - Add Greater Predictability
    - Essentially the Same Information as Part 50
    - Combines Construction Permit & Operating License with Conditions for Plant Operation – Combined License (COL)
    - Specify Applicant Inspection, Tests, Analysis and Acceptance Criteria (ITAAC)

An EP Perspective on New Reactor Licensing

Regulatory Process *(cont'd.)*

- **10 CFR Part 52 – Alternative licensing process**
  - **Acceptance Criteria**
    - Provide Reasonable Assurance that the facility has been constructed and will operate in conformity with the license and applicable regulations

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- **10 CFR Part 52 – Combined License**
  - **NRC**
    - Authorize fuel load ONLY after ITAAC met
    - Periodic Federal Register Notice as ITAAC met
    - 180 days prior to scheduled initial loading of fuel
      - *Publish notice of intended operation in Federal Register*
    - Hearing opportunity if petitioner demonstrates that Acceptance Criteria not met

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- COL
  - **10 CFR 52.79(d)** ***proposed 10 CFR 52.79(22)***
    - Obtain Certifications from agencies with EP responsibilities that:
      - (A) Plans are practicable
      - (B) Commitment to further develop plans including field demonstrations
      - (C) Commitment to execute responsibilities

Section 13.3, "EP" of the SRP and DG-1145

<u>Regulatory Process *(cont'd.)*</u>

- COL
  - ***Proposed 10 CFR 50.54(gg)***
    - Allows operation up to 5% power with offsite deficiencies
    - Much like existing requirement in 10 CFR 50.47(d)

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- 10 CFR Part 52 – Combined License
  - COL can incorporate by reference
    - Design Certification
    - Early Site Permit
  - Issues resolved in ESP or Design Certification are precluded from reconsideration at COL Stage

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- Standard Design Certification
  - ***10 CFR 52 Subpart B***
    - Allows certification of Nuclear power facilities separate from filing an application for construction or combined license
      - No specific EP Requirements

Section 13.3, “EP” of the SRP and DG-1145

Regulatory Process *(cont’d.)*

- Early Site Permit (ESP)
  - **Independent of Plant Design**
  - **Valid for 10 – 20 Years, Renewable**
  - **Resolve early issues on**
    - Site Safety
    - Emergency Preparedness
    - Environmental Protection

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- **Early Site Permit (ESP)**
  - ***10 CFR 52.17***
    - (b)(1) Unique Physical Characteristics that could pose significant impediment to developing EP
    - (b)(2)(i) Major Features (NUREG-0654, Supplement 2)
    - (b)(2)(ii) Complete and Integrated Plans

Section 13.3, "EP" of the SRP and DG-1145

Regulatory Process *(cont'd.)*

- ESP *(cont'd.)*
  - ***10 CFR 52.17*** *(cont'd.)*
    - (b)(3) Describe contacts and arrangements with agencies with EP responsibilities [(b)(1) & (b)(2)(i)], ***OR***
    - Obtain Certifications from agencies with EP responsibilities that [(b)(2)(ii)]:
      - (3)(i) Plans are practicable
      - (3)(ii) Commitment to further develop plans including field demonstrations
      - (3)(iii) Commitment to execute responsibilities

Section 13.3, "EP" of the SRP and DG-1145

- Standard Review Plan (SRP) (NUREG-0800) Section 13.3, "Emergency Planning"

- COL Application Guidance (DG-1145) Section 13.3, "Emergency Planning"

Section 13.3, "EP" of the SRP and DG-1145

SRP Section 13.3, EP

- Provides for review of EP in
  - Construction Permit (CP)
  - Operating License (OL)
  - Early Site Permit (ESP)
  - Standard Design Certification (DC)
  - Combined License (COL)

Section 13.3, "EP" of the SRP and DG-1145

SRP Section 13.3, EP

- Identifies Review Interfaces within SRP
- Identifies Regulatory Requirements
- Establishes Acceptance Criteria to existing Regulatory Guidance
- Provides Technical Rationale
- Outlines Review Procedure
- Proposes generic Evaluation findings
- Extensive Reference list
- Generic EP ITAAC Table

Section 13.3, "EP" of the SRP and DG-1145

SRP Section 13.3, EP

- Consideration of existing programs
  - Is it applicable to proposed reactor
  - Is it up-to-date
  - Reflects and incorporates new reactor

Section 13.3, "EP" of the SRP and DG-1145

SRP Section 13.3, EP

- Emergency Action Levels (EALs)
  - NEI 99-01applicable EALs used
  - NEI 99-01 EAL development guidance
  - Passive reactor designs EALs

Section 13.3, "EP" of the SRP and DG-1145

SRP Section 13.3, EP

- Inspection, Test, Analysis, Acceptance Criteria (ITAAC)
  - Generic EP ITAAC provided in Table 13.3.1
  - Develop with Industry & public participation
  - Based on existing NUREG-0654 criteria
  - Not all-inclusive, or exclusive
  - Applicant proposes and accomplishes
  - Case-by-case determination

Section 13.3, "EP" of the SRP and DG-1145

SRP Section 13.3, EP

- Offsite EP Guidance
  - Current REP-series guidance documents
  - Associated Memoranda
  - Radiological Emergency Preparedness: Planning Guidance, February 28, 2003

Section 13.3, "EP" of the SRP and DG-1145

SRP Section 13.3, EP

- Standard Design Certification EP (not required)
  - EP features are technically relevant to the design, and not site-specific, and usable for a multiple number of units or sites
  - Programmatic aspects of EP are COL applicants' responsibility
  - Facilities, functions, and equipment to support EP
    - TSC, OSC, Personnel Decontamination
      - Location, size, habitability, ventilation systems
      - ERDS, SPDS, Voice and data Communications

Section 13.3, "EP" of the SRP and DG-1145

DG-1145 Section 13.3, EP

- Provides guidance on EP information in a Combined License for a
  - Custom design
  - Certified Design
  - Certified Design with ESP

Section 13.3, "EP" of the SRP and DG-1145

DG-1145 Section 13.3, EP

- Addresses EP information in a Combined License
  - Application & Emergency Plan Content
  - Multi-Unit Site considerations
  - EP ITAAC

Section 13.3, "EP" of the SRP and DG-1145

ACRS Preliminary Questions

- Substantive change to Section 13.3 is incorporation of Part 52 process
  - EP ITAAC
  - "Predictive" reasonable assurance finding
  - Timing of exercise

Section 13.3, "EP" of the SRP and DG-1145

ACRS Preliminary Questions

- Guidance for "green-field" sites
  - Existing guidance is applicable
  - Considered in development of generic EP ITAAC
  - Continue discussion with DHS
  - Plans needed at COL application stage
  - Implementing Procedures developed later

Section 13.3, "EP" of the SRP and DG-1145

ACRS Preliminary Questions

- Completeness of EP ITAAC Table for ESP
  - Generic EP ITAAC provided in Table 13.3.1
  - Develop with Industry & public participation
  - Based on existing NUREG-0654 criteria
  - Not all-inclusive, or exclusive
  - Applicant proposes and accomplishes
  - Case-by-case determination

Section 13.3, "EP" of the SRP and DG-1145

Questions?

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NUREG-0800 Section 13.3 "Emergency Planning"

Alan Nelson

Director Emergency Preparedness

NUCLEAR ENERGY INSTITUTE

ALWR EP Task Force

- Task Force Members Represent
  - Constellation
  - Dominion
  - Duke
  - Entergy
  - Exelon
  - Progress Energy
  - SCANA
  - Southern Nuclear
  - STP
  - TVA
- Representing Reactor Types
  - ABWR
  - AP1000
  - ESBWR
  - U.S. EPR

NEI

Task Force Projects

- Emergency Action Levels (EALs) for passive reactors
- Review and comments on NRC Draft Documents
  - DG 1145
  - Standard Review Plan

NEI

Emergency Action Levels

- NEI 99-01, "Methodology for Development of Emergency Action Levels" Endorsed in RG 1.101
  - Template for existing fleet
  - Currently used by over 70% of existing fleet
  - 2007 to 2008 - 100%
  - EALs
    - Radiological events
    - Cold shutdown events
    - Security
    - Hazards
    - Fission product barrier integrity
    - System malfunctions

NEI

Emergency Action Levels

- NEI 07-01, "Methodology for Development of Emergency Action Levels for Advanced Passive Light Water Reactors"
  - AP1000 and ESBWR
  - Adapts NEI 99-01 method to new passive reactors
  - Industry and vendor development phase

NEI

Standard Review Plan NUREG 0800

- Section 13.3 "Emergency Planning"
- NRC requested public comments
  - Federal Register, September 29th
  - Comments by November 13
  - Comments submitted by NEI November 9th
- ALWR EP Task Force
  - Draft document review and comments

NEI

Standard Review Plan
NUREG 0800

- Pleased with level of detail provided by NRC in document
- Following concerns:
  - New reactor at existing site opening review of existing emergency plan
  - Expansion of original agreed on ITAAC
  - Generic communications referenced
  - Requirement to submit off site procedures

NEI

Standard Review Plan
NUREG 0800

- Significant concern:
  - Document provides guidance on NRC and FEMA review process
    - Detailed guidance on NRC review
    - Limited guidance on expectations for FEMA review

NEI