SEQUOYAH INFORMATION NEEDS

Dominant Sequence Selection

Background

The NRC staff has selected dominant sequences / sequence groupings for internal and external events (Refer to Attachment 1), as well as the availabilities of systems that may impact post-core damage containment accident progression, containment failure, and radionuclide release for Surry to be used in the SOAR-CA.

The dominant internal event sequence/sequence groups have been identified using the plant-specific, Level 1 Standardized Plant Analysis Risk (SPAR) models, Version 3.31[1]. A screening threshold, based on core damage frequency (CDF), was used as a guide to select the dominant sequence/sequence groups. Generally, sequence/sequence groups with CDFs less than 10^{-6} (10^{-7} for sequence/sequence groups involving containment bypass) were eliminated from further evaluation. However, the number of dominant sequence/sequence groups, along with other factors such as licensee PRA results and impacts of external events may also affect the final sequence/sequence group selection.

In addition to the internal events, an assessment of the dominant external event sequences was performed. Generic information sources, along with plant-specific data sources (e.g., IPE, IPEEE) and the external event SPAR model developed for Surry were used to select dominant external event sequences (based on CDF estimates) due to internal fires, internal and external flooding, seismic events, and severe weather (e.g., tornados).

Containment systems analyses were performed by developing system dependency table showing support systems required for performance of targeted front line systems. Using these dependency tables and available system models (e.g., fault trees), the availability of front line systems with the potential to impact post-core damage containment accident progression, containment failure and radionuclide release were determined.

Information Need

Summaries of the dominant internal and external sequences, and the containment systems states identified by the NRC are being provided (Attachment 1) for your review. Compare these results with the latest site-specific PRA and identify any major discrepancies or omissions in dominant sequences / sequence groupings for internal and external events, as well as the availabilities of systems that may impact post-core

1 The Level 1 SPAR Model for Surry(Version 3.31) used for this analyses has been subjected to a dominant cutset level review against licensee PRA results. Modeling and data differences that were discovered during this review have been documented and are being addressed in the SPAR Model Development Program.

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damage containment accident progression, containment failure, and radionuclide release for Surry in preparation for discussion during our visit to your facility.

Emergency Preparedness Information Needs

Background

Emergency response programs are designed to protect public health and safety during radiological accidents and will be modeled as realistically as practical for the SOAR-CA. This will be accomplished by estimating evacuation speed and timing for various population cohorts as well as modeling other aspects of emergency response such as sheltering and precautionary protective actions. The implementation of precautionary protective actions will be modeled when the scenario indicates escalation through emergency classes and the state/local plan allows/directs such implementation. Precautionary actions are generally implemented at the Site Area Emergency level, but may be considered at the Alert. Declaration of emergencies will modeled based on emergency action levels (EALs) reached during the scenario. Actual control room readings will not be available from computer runs but basic plant parameters are available and will allow estimation of many EAL values. It will be assumed that emergency plans will be implemented by offsite authorities; emergency workers will perform the tasks assigned to them; and the public will largely obey directions form officials.

Information Need

The information needed to model the emergency response program includes the following:

1. Complete Evacuation Time Estimate Report.

   Provide the full Evacuation Time Estimate (ETE), not just a summary. The ETE usually provides an estimate of the time for total evacuation of an area and estimates of population preparation times. The licensee is not being asked to perform any analyses regarding population cohorts, only to provide the most recent full ETE report.

2. Emergency Classification Procedure and Operator Aid Used for Classification in the Control Room.

   Precautionary protective actions will be modeled based on declaration of emergency classes and implementation of offsite plans. The procedure used for classification of emergencies will be used to estimate when, during a given scenario, an emergency class would be declared.

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

a) The chapter or procedure that is used by decision makers to guide their decision on implementation of public protective actions during actual emergencies.

b) The chapters, appendices or procedures that address protective actions for special needs populations and protective actions for schools.

4. Full size (color if available) evacuation route map and evacuation travel directions for the public in the various emergency planning areas within the 10-mile plume exposure pathway emergency planning zone, identifying evacuation routes and travel directions to facilitate evacuation modeling.

Mitigative Measures Analysis Information

Background

Over the past 25 years, the industry has significantly improved plant design and operation that focused on improved plant performance and event mitigation. The SOARCA will consider these improvements and credit any existing and committed-to mitigative measures that can delay or prevent core damage or containment failure. The staff will assume that basic engineer systems and emergency operating procedures were credited in the licensee's IPE and in the staff's enhanced SPAR models and will focus this effort on SAMGs, EDMG's, and any additional new mitigative measures identified as part of this effort.

Information Need

In an effort to credit mitigative measures that are in place and that are being implemented as a result of recent efforts (e.g., B.5.b), the staff will need the following information to assess the availability, effectiveness, and timeliness of these actions:

1. Procedures - Severe Accident Mitigation Guidelines, Extreme Damage Mitigation Guidelines, and any additional procedure or guidance that controls the order / priority and timing for the implementation of mitigative measures (electronically, if available).
2. Based on the event sequences of interest provided by the NRC, identify the applicable and available (including primary and support systems availability) mitigative measures, the anticipated start time for implementation (consider the order and times for implementing other mitigative measures from EOPs, SAMGs, and EDMGs), the time to complete implementation, the bases for these times, and the anticipated effects from implementing these measures (e.g., flow rates, available volumes, etc.).

Structural Information

Background

As part of a consequence analysis it is important to understand the containment response to accident conditions. The NRC will perform site-specific containment analysis to better understand expected timing and failure modes under severe accident conditions.

Information Need

General Containment Data

1. Provide steel containment vessel construction/fabrication drawings - including detailed dimensions, and copies of stress analysis reports,

2. Provide drawings, and stress analysis/failure testing of penetration bellows, personnel locks and equipment hatches.

3. Provide details of first supports/hangers, and routing of penetrating piping from the steel containment structure. Especially, penetrations at higher elevations.

4. Provide concrete shield building structure construction/fabrication drawings – including detailed dimensions, and copies of stress analysis reports.

5. Provide details of any reported degradation(s) in steel containment vessel and concrete shield building structures.

Containment Data for Input into MELCOR

1. Description of containment failure criteria

2. Chemical composition and rebar content of concrete of the reactor cavity

Power History and Fission Product Inventory Analyses

1. Provide both axial and radial fuel geometry.

2. Provide complete fuel material specification relating to fuel enrichment, any burnable absorbers, and cladding material.

3. Provide reactor design information including the number and type of assemblies, reactor power, assembly layout, the typical boron letdown curve, and the typical cycle burnup.

MACCS Information

Background

MACCS2 computer code is undergoing a number of improvements to provide more realistic results including but not limited to the following: increasing of angular resolution from 16 sectors to any number of sectors, updating the plume meander model, increasing the number of evacuation cohorts, and other code performance improvements.

Information Need

To effectively utilize the MACCS2 code improvements to provide the most realistic results, the following information is needed for MACCS2 modeling:

1. Reg Guide 1.23, "Onsite Meteorological Programs," data that includes 8760 consecutive hours of raw meteorological data provided in electronic format, if available.
2 Although not required by Regulatory Guide 1.23, the staff is requesting the following additional information if available: hourly precipitation data from the same one year (or any part of) time period as the raw meteorological data; most recent evaluation of seasonal (winter, spring, summer, and autumn) and diurnal (day versus night) mixing heights; and breathing and shine shielding factors for the average building types in the area near the plant.

MELCOR Information

Background

NRC has a MELCOR model of the Surry 3-loop Westinghouse pressurized water reactor with a sub-atmospheric containment. The existing model requires an update to accurately represent the current fuel and fission product inventory. The Surry model was recently used in an unrelated study that resulted in the development of a model of the auxiliary building. Therefore, the information being requested for Surry is relatively small, and will be used to (a) confirm or request critical design parameters, (b) improve the accident response models, and (c) obtain the most recent fuel burn-up data for fission product inventory calculations.

General Plant Data

Current version of the UFSAR including all tables and figures

PRA Severe Accident Insights

1) Realistic estimated duration of station batteries during total ac power loss (with and without load shedding)
2) Pump-seal leakage characteristics (gpm following loss-of-seal cooling flow, gpm with seal failure, criteria for seal failure)

3) Pressurizer and SG valve failure characteristics (e.g., % likely to fail after x cycles, fails above xxx °F)
4) Containment failure characteristics (pressure criteria, pressure/temperature criteria, leak location(s), and leakage area(s))
5) Water level calculation for pool depths during accident conditions

    a) Wet or dry reactor cavity for sequences such as RCS LOCAs and station blackout. At what water elevation will water flow between the cavity and the basement?
    b) Under what conditions (if any) could the outside surface of the vessel lower head be flooded? (e.g., full RWST dump + RCS water + ice melt)
    c) Characteristics of the ice melt rate on a loss-of-refrigeration

6) Unique plant-specific mitigation features that should considered in the severe accident analysis.

Normal Operating Conditions

a. Pressurizer and SG pressure
b. Reactor coolant system (RCS), core, and core bypass flows
c. Characterization of vessel leakage flows (e.g., downcomer to upper head, hot leg nozzle to downcomer, core bypass)
d. Hot and cold leg temperatures
e. Core pressure drop, vessel pressure drop, and pump head
f. Best-estimate heat loss
g. Feedwater temperature and flowrate
h. Steam flowrate + any SG blowdown flow
i. SG secondary water mass
j. SG secondary recirculation rate
k. Containment pressure and temperature
l. Initial ice mass and temperature

Plant-Specific MAAP (or equivalent) thermal-hydraulic model and supporting documentation

Reactor Vessel, RCS, and Steam Generator (SG) Data

1. Vessel

    a. Vessel ID, wall thickness, height, and layout of internals (elevation drawings)
    b. Core shroud dimensions and former dimensions (and mass)
    c. Lower vessel head radius of curvature
    d. Lower vessel head wall thickness

e. Hot leg nozzle material construction, thicknesses, geometry (drawing), and mass

2. RCS

Hot and cold leg ID, lengths, wall thicknesses, and layout(drawing)

1. Steam generator

Number of tubes and number plugged

   i. Active heat transfer area
   ii. Inactive heat transfer area

2. Steam generator

Number of tubes and number plugged

   i. Active heat transfer area
   ii. Inactive heat transfer area

3. Auxiliary feedwater pumps

   a. Activation logic
   b. Types and number (turbine-, and motor-driven)
   c. Injection flow
   d. Steam flow for turbine-driven pumps
   e. Activation logic. How is it controlled in a SBO (e.g., to a specific level? Is there sufficient instrumentation?)
   f. What happens to AFW when dc power fails (e.g., continues to run at last steam valve setting)?
   g. Source of water and the tank capacities (CST and hotwell?)

4. Steam lines

   a. Piping ID and wall thickness
   b. Flow restriction area
   c. Length to isolation valve
   d. Relief valves

      i. Number, types, and power dependency
      ii. Flowrate
      iii. Opening/closing setpoints

Fuel Data for 3 Consecutive Cycles

a. Fuel type (Vendor/model)
b. Total cycle burn-up
c. Cycle power history and shutdown time
d. Fuel loading pattern
e. Location and construction of integral burnable poison rods
f. Fuel assembly enrichment and metric ton of uranium (MTU)
g. Fuel assembly average power
h. Beginning of cycle, middle of cycle, and end of cycle burn-ups for each
i. Average boron concentration during the cycle
j. Average axial power profile (assembly-specific)

Note: We have (a) core loading pattern and (b) assembly power and burnup maps for Unit 1, Cycle 14; and Unit 2, Cycle 12 and Cycle 13

Containment

1. Drawings or data detailing,

   a. Layout and vertical cross-section drawings of the containment
   b. Nominal leakage (%vol/day)

2. Reactor water storage tank water switchover criteria, if different than from ECCS

3. Reactor cavity concrete chemical composition and rebar content

4. Recirculation fans

   a. Activation logic
   b. Flow rate

5. Leakage between upper and lower compartments