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N Reactor External Events Probabilistic Risk Assessment

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ABSTRACT

An external events probabilistic risk assessment of the N Reactor has been completed. The methods used are those currently being proposed for external events analysis in NUREG-1150. Results are presented for the external hazards that survived preliminary screening. They are earthquake, fire, and external flood. Core damage frequencies for these hazards are shown to be comparable to those for commercial pressurized water reactors. Dominant fire sequences are described and related to 10 CFR 50, Appendix R design requirements. Potential remedial measures that reduce fire core damage risk are described including modifications to fire protection systems, procedure changes, and addition of new administrative controls. Dominant seismic sequences are described. The effect of non-safety support system dependencies on seismic risk is presented.

INTRODUCTION

The 1982 Triennial Review [1] recommended that a probabilistic risk assessment (PRA) should be performed on the N Reactor, the only large water-cooled, graphite-moderated reactor in operation in the United States. In May 1986, a Level 1 PRA was initiated. After the Chernobyl incident, the scope of the study was expanded to include Level 2/3 studies with treatment of external events. Results of the N Reactor Level 1 study were published in August 1988 [2]. Core damage frequency was estimated as 6.4 E-05/yr. Sandia National Laboratories was selected to conduct the N Reactor external events risk assessment under subcontract to Westinghouse Hanford Company in late 1987. They were chosen because of their recognized expertise in this area. Objectives of the contract were to (1) evaluate the contribution of external events to plant core damage frequency, (2) identify plant modifications to reduce expected risk of operation if needed, (3) furnish input for the Level 2/3 PRA, and (4) transfer technology of external events PRA methodology to Westinghouse Hanford Company.

The N Reactor external events analyses were started in January

1988 and completed in September 1989. Four types of external hazards survived screening based on a mean rejection frequency of 1 E-06/yr. They were earthquake, fire, extreme winds and tornadoes, and external flooding. Detailed risk assessments were conducted for earthquake and fire hazards, while bounding analyses were used to evaluate risks from extreme winds and external flooding. This paper presents initial results and a brief discussion of engineering insights gained from the study.

METHODOLOGY

The PRA procedures used for the N Reactor analyses are based on the following general concepts:

1. External events analyses are based on the internal event risk assessment plant system models and fault trees.

2. Systematic screening is used to evaluate all external events to which the plant might be exposed and eliminate unimportant events.

3. Evaluation of similar events is coordinated to avoid duplication of effort and minimize data-gathering efforts.

4. Computer-aided screening techniques and generic failure data are used before detailed component failure analyses to minimize effort on failure analyses.

Procedures based on these concepts have been applied (in whole or in part) to six power plants as part of the U.S. NRC-sponsored Unresolved Safety Issue A-45 resolution program [3], to the Peach Bottom and Surrey power plants as part of NUREG-1150 (draft 1989) [4] and to the N Reactor as reported here. A full description of these procedures is given in Bohn and Lambright (draft 1988) [5].

FACILITY DESCRIPTION

A plot plan of the N Reactor is shown in Figure 1. The buildings of principal importance are 105-N, 109-N, 181-N, 182-N and 184-N.

The reactor and its cooling systems are housed in two adjoining buildings, 105-N and 109-N. The reactor, including its control and trip systems, the control room, and cable spreading room are located in 105-N. Primary coolant pumps, steam generators, the pressurizer, graphite shield cooling components, and other auxiliaries are located in 109-N.

Located within 105-N, the reactor itself consists of a horizontal array of 1,003 Zircaloy process tubes penetrating a 1,800 ton graphite cuboid approximately 33 ft by 33 ft at the face and 39 ft long. Interlocking graphite bars make up the cuboid. A composite steel and high-density concrete thermal shield box structure surrounds the core. Both the core and thermal shield are supported by the reactor pedestal, which is a separate structure from the 105-N building.

The river pump house (181-N) is located at a lower elevation adjacent to the Columbia River. Electrical pumps supply the circulating raw water system (CRW) during normal operation. Three diesel-powered, emergency core cooling (ECCS) low-lift pumps are also housed in this structure.

The 182-N Building houses five diesel-driven pumps servicing the ECCS and confinement fog

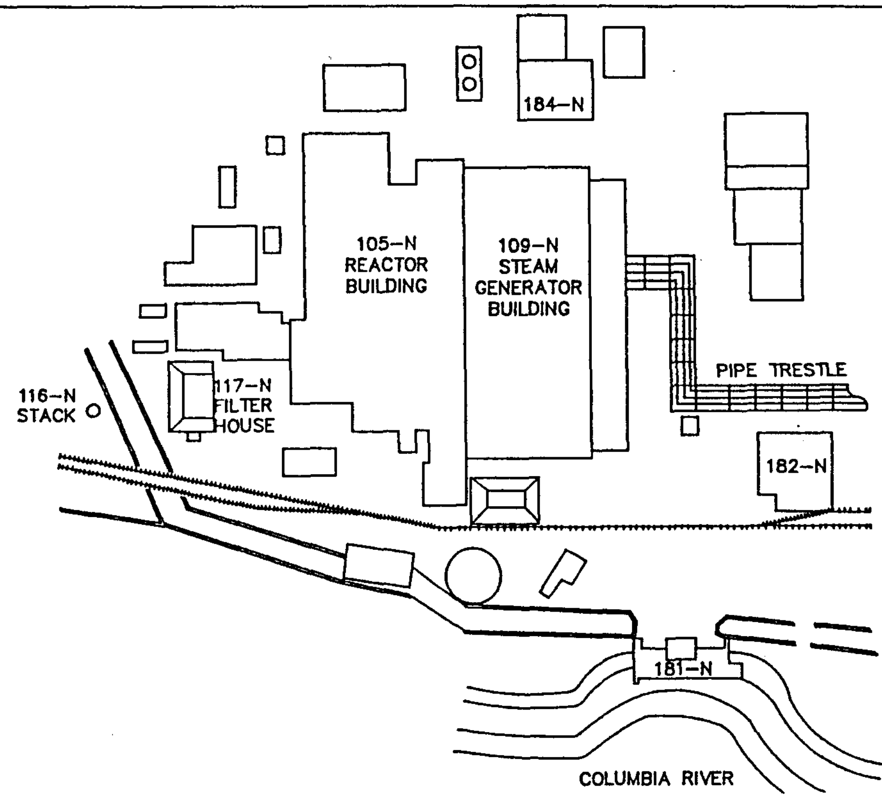

Figure 1. N Reactor Plot Plan.

sprays. High-pressure injection pumps and both low- and high-pressure auxiliary water systems are also located in the basement of 182-N.

WIND BOUNDING ANALYSIS

N Reactor was evaluated for risk resulting from tornadoes and straight winds. Hurricanes were not considered because of the plant's inland location separated the ocean by a high mountain range.

The reactor site is located in region III of the U.S. NRC tornado risk regionalization scheme [6]. An alternative regionalization scheme by Twisdale and Dunn [7] places the plant in region D of a four-region scheme. Thus, the site has the lowest tornado occurrence rate in the continental United States. Site-specific studies, when extrapolated to the N Reactor site, indicate that the probability of a maximum wind speed of 175 mi/h is around 1 E-07/yr. At higher wind speeds tornado winds govern design over straight wind for this site.

Wind design loads for the original plant design were based on a maximum straight wind load of 78 mi/h at a height of 50 ft above ground level in accordance with the 1961 Uniform Building Code.
Plant standards were revised in 1974 to include a design basis tornado with a maximum speed of 175 mi/h. Critical reactor facilities have been evaluated against the design basis tornado and were either accepted or upgraded in the interval since 1974.

During a design basis tornado both onsite and offsite power will be lost resulting in a demand for the ECCS. Critical facilities required to provide a success path for ECCS are shown in Figure 1. These include 105-N (the building housing the main control room), 182-N (high-lift pump house), 181-N (river pump house), and 109-N (heat exchanger building). The ECCS system was upgraded and qualified for both design basis earthquake and tornado as part of a recent seismic safety enhancement program. Provision of the success path is dependent on completion of several minor tornado resistance upgrades to buildings 181-N and 182-N before facility restart.

Assuming completion of the upgrades, the tornado risk at N Reactor is considered acceptable. Results of the tornado risk assessment support the original decision to qualify the ECCS system.

FIRE RISK ASSESSMENT

The N Reactor was authorized in 1958 and achieved its design thermal power rating in 1964; thus, its design and construction predate many contemporary requirements.

After the Browns Ferry fire additional fire protection requirements were imposed on existing commercial nuclear power facilities. These requirements are specified in 10 CFR 50, Appendix R. In particular, separation requirements are specified for independent trains of safety-related equipment. Modifications have been made over the years at the N Reactor to bring portions of the safety related systems into compliance with Appendix R.

Post-Chernobyl safety reviews identified physical separation of redundant ECCS diesel pump trains in buildings 181-N and 182-N as new Appendix R concerns. These concerns stemmed from walkdown notes by various reviewers. In response to these concerns, plant modifications were made. These included erection of new fire walls between adjacent diesel pumps and re-routing of power and control cabling to provide physical separation between ECCS diesel trains in the 181-N and 182-N buildings. As with previous upgrades, the decisions were based on qualitative "engineering judgement" rather than a quantitative decision making process.

Plant Vulnerabilities and Engineering Fixes

The total fire-induced core damage frequency computed for the N Reactor is 1.8 E-04/yr. Transients dominate the accident sequences as shown in Table 1.

Table 1: Fire accident sequences with annual core damage frequencies

Sequence	Fire Area	Mean Core Damage Freq./yr
T8	109-N Access Corridor and 109-NT Bsmnt.	1.3 E-04
	Main Control Room	3.2 E-05
	105-N Access Corridor	4.6 E-05
	105-N Cable Spreading Rm.	6.6 E-07
T4	Bldg 184 Cable Runs (Plant Air)	1.3 E-06

Sequence T8 results from an early failure of the primary/secondary cooling systems (PCS/SCS) caused by a transient initiating event and early failure of ECCS. Sequence T4 results from an early failure of the PCS/SCS caused by an initiating event and subsequent failure of ECCS flow in one of 16 risers (partial core involvement). Over 98% of this frequency is attributable to a "pinch point" in control and power wiring where it exits the cable spreading room. The general arrangement of this area is shown in Figure 2.

compliance with a level of protection called "improved risk" as used by the insurance industry.

<u>Cable Trays in Building 109</u>

The 12 cable trays identified in Figure 2 contain control and power wiring for many plant sytems. These include the primary cooling system pumps and high-pressure injection, actuation, and power circuits for the ECCS system, controls for the confinement fog spray system, the plant raw cooling water

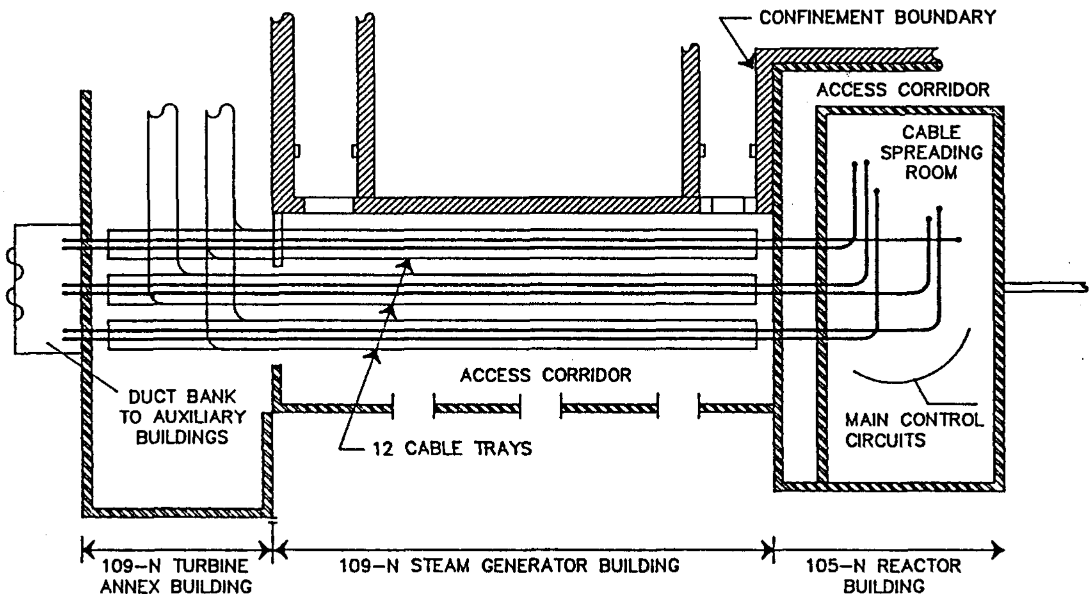

Figure 2. Power and Control Wiring "Pinch Point."

The cable trays and conduit in this area are the original installation with limited additions and modifications which have taken place over the years. There is no separation between control and power wiring for separate trains of cooling systems.

All areas in Figure 2 have existing fire protection systems. Fire protection systems at N Reactor are designed to provide

supply, portions of the secondary cooling system, and communications circuits from the main control room to control rooms in the pump houses and the local power plant in 184-N.

Both the access corridor and the Turbine Annex basement area are large, open areas. Cabling in this area has a "Flamastic" coating so self-ignited fires from "hot-shorts" were not considered. Transient combustibles are the main initiators for this area. Small fires do not damage the trays in both areas. Large fires can damage the trays anywhere in the corridor in 2 to 6 min, based on COMPBRN simulation runs. Similar results were obtained in the basement area immediately surrounding the trays.

The cable trays are protected by automatic sprinklers (wet standpipe system) below and above the trays. Lower sprinkler heads are spaced approximately 10 ft apart, centered below the middle tray. The upper run is centered over the middle tray run with heads on 20-ft centers. All sprinklers are equipped with fusible link heads. Suppression credit was given for the sprinklers below the trays if any point of the fire pool was within 1 meter radially of a sprinkler head. This represents 50% of the floor area immediately below the trays. Fires at other locations damage the trays and wiring before the sprinklers respond.

Similar arguments apply to those cable tray sections passing through the building 105-N access corridor and in the immediate area surrounding the cable tray penetrations of the cable spreading room. Short times to damage indicate a need for better fire detection and suppression in these areas.

Main Control Room

Control rooms at the N Reactor are continually manned and equipped with adequately designed and maintained halon systems. Scenarios are based on suppression of 99 out of 100 fires before control wiring is damaged. The remaining fire scenario assumes control room abandonment because of smoke from a cabinet fire. Although remote shut-down capability exists at the N Reactor, the control room procedures are not yet approved and in place. Assuming that the procedures are implemented as part of a restart effort, recovery will reduce this frequency by a factor of two.

Risk Reduction Measures

For the 109-N access corridor and 109-NT basement area modifications are proposed that will provide for earlier detection and suppression because of the very short time to damage. Early warning smoke detectors installed in accordance with National Fire Protection Agency standards and tied into the existing control panels and alarm system would provide adequate detection. Alternate automatic actuation circuitry for the existing wet standpipe sprinkler system may be considered to improve suppression times. Additional fire protection blankets for passive protection may be an alternative to sprinkler modifications.

The access corridor in 105-N is a dead-end, low-traffic area. Administrative controls may be implemented to make this area a flammable-free area. This would imply a full-time fire watch for future plant maintenance in this area.

Assuming that these modifications are properly implemented, the mean frequencies for the two most dominant sequences would be reduced by a factor of from 10 to 20. Coupled with development of additional control room procedures, the overall fire risk core damage frequency can be easily reduced to values in the neighborhood of 5 E-05/yr; comparable to commercial nuclear power plants.

SEISMIC RISK ASSESSMENT

Background

A seismic upgrade program has been conducted at the N Reactor in preparation for restart at the same time as the external events PRA. Elements of the upgrade program were developed to respond to post-Chernobyl review comments on the seismic qualification of the N Reactor ECCS and graphite shield cooling systems (GSCS). Commitments were made to qualify the ECCS cooling system and the necessary supporting systems for seismic and wind environmental conditions (i.e., safe shutdown earthquake and design basis tornado). This resulted in a $24 million upgrade program with 212 category I fixes, analysis of 1480 seismic III/I problems, and 530 III/I fixes.

Walkdowns and plant modifications stemming from the seismic upgrades program have influenced the PRA results. There are no instances of localized failure in non-safety class equipment or building partitions (seismic III/I failures) that would lead to dominant core damage sequences.

Initiating Events

Seismic risk assessment of the N Reactor is based on the same event trees developed for the internal events analysis of the plant. Initiating events considered include the following:

- Process tube rupture (ECCS ineffective)
- Large LOCA
- Small LOCA
- Building 182 failure
- Transient Type 1 (LOSP)
- General transient (PCS initially available)

The initiating events are listed in a hierarchy from most severe to least least severe. Events in the hierarchy above an initiating event are precluded from the accident sequence for that event. The sum of initiating event probabilities must be one for each increment of the seismic hazard curve.

Dominant Accident Sequences

Initial seismic screening reduced the number of sequences to 13. Subsequent quantification with best estimate random failures means and best estimate seismic fragilities and responses identified five dominant sequences:

- Building 182 (45%)
- LOSP T8 (21%)
- PTR-1 (19%)
- SLOCA T8 (10%)
- TRANS T8 (2%)

Percentage contributions are based on a Monte Carlo uncertainty analysis. The total mean core damage frequency was determined to be 6.7 E-05/yr. The range factor (defined as the ratio of the 95th percentile to the 50th percentile of the distribution) on total core damage frequency was found to be 35.

Description of Accident Sequences

N Reactor has three cooling systems available to mitigate accidents. System G (PCS/SCS), System A (ECCS), and System C (GSCS).

The dominant accident sequence results from the failure of building 182. Building failure leads to accident sequence GA-C. No other failures are required to cause this transient accident sequence. Building collapse is assumed to fail all PCS/SCS support pumps and all of the ECCS high-lift diesel pumps. GSCS does not fail because its pumps are located elsewhere.

The next most important sequence is process tube rupture. This arises at high ground-motion input levels (0.75 g) when the primary shield wall of the reactor starts to fail. This failure takes the form of uplift and rocking of the primary shield wall, which leads to shear failure of the process tubes at the inlet and outlet face of the primary shield. No other failures are required to cause this accident sequence. This sequence is similar to reactor vessel rupture sequences reported in NUREG-1150.

The next three sequences in order of importance are TRANS T8, SLOCA T8, and LOSP T8. All result from the same logical combination of cutsets and component failures. In each case, the sequences involve GA-C system failures. The dominant cutsets consist of failure of the ECCS silo structure, which fails the ECCS system in combination with electrical bus failures (both 13.8 kV and 4 kV) which fail PCS/SCS. As before, GSCS is not failed. The PCS/SCS failures are caused by dependency of instrument air on 4-kV busses.

Basic Event Importance

The mean seismic core damage frequency for the N Reactor is comparable to that for two commercial plants, Surry and Peach Bottom, as reported in NUREG-1150 (draft 1989) [4]. Results of the analysis have not been available for sufficient time to evaluate potential risk reduction measures. Basic event importance to the mean values was determined by setting the seismic failure probability to zero for each component and recalculating the mean point estimate. Risk reduction potentials are:

o ECCS silo	37.6%
o Building 182	19.4%
o 13.8-kV Bus	5.7%
o 4-kV Bus	11.9%
o Primary shield wall	1.4%
o Ceramic insulators	1.3%
o Building 105	0.4%

Building 182 and the ECCS silo structure were both seismically upgraded to qualify for a 0.25-g SSE earthquake in recent restart programs. This has influenced the results of the current external events analysis.

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