SEISMICALLY INDUCED ACCIDENT SEQUENCE ANALYSIS OF THE ADVANCED TEST REACTOR

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ABSTRACT

A seismic probabilistic risk assessment (PRA) was performed for the Department of Energy (DOE) Advanced Test Reactor (ATR) as part of the external events analysis. The risk from seismic events to the fuel in the core and in the fuel storage canal was evaluated.

The key elements of this paper are the integration of seismically induced internal flood and internal fire, and the modeling of human error rates as a function of the magnitude of earthquake. The systems analysis was performed by EG&G Idaho, Inc. and the fragility analysis and quantification were performed by EQE International, Inc. (EQE).

INTRODUCTION

EG&G Idaho has been performing the external events analysis as part of the Level 1 PRA for the ATR. The seismic analysis is an integral part of the external event and spatially dependent accident analysis of the ATR PRA. This paper describes the integration of seismically-induced internal flood and internal fire in the seismic risk analysis portion of the PRA performed for the ATR. The human error rates also were explicitly modeled as a function of the magnitude of earthquake.

In most modern PRAs for U. S. commercial nuclear power plants, the impact on core damage frequency from seismically induced internal fires or floods, and from seismically induced leakage or rupture in the fuel storage pool, has not been evaluated. For the ATR, the impact of seismically induced flood and fire on the safety components was modeled explicitly. The ATR PRA program was sponsored by the DOE under contract No. DE-AC07-76ID01570.

DESCRIPTION OF THE PLANT

The ATR is a 250-MW_{th} test facility located at the Idaho National Engineering Laboratory (INEL). The ATR, which began operation in 1968, has a smaller core, higher power density, lower primary coolant system (PCS) pressure and temperature (350 psig and 170°F), and greater ratio of coolant weight to power, than typical commercial pressurized water reactors.

Designed to study the effects of intense irradiation on samples of reactor materials, the unique cloverleaf shape of ATR's 1.2–m high core provides positions for nine in–pile tubes (flux trap positions), and numerous smaller irradiation locations. The lobes of the cloverleaf core allow various power levels to be established at different lobe positions. Separate loop systems for each in–pile tube provide coolant at the experiment's designated temperature, pressure, and flow rate. A comparison of the ATR to a typical commercial pressurized water reactor (PWR) is presented in Table 1.

The top of the ATR reactor vessel is located at ground level, and two floors below house PCS pumps and heat exchangers, switchgear, loop systems, and other equipment. Framed with structural steel, the confinement structure above the reactor is designed as a barrier to radionuclide release into the atmosphere.

The ATR was originally designed to 1960 Uniform Building Code (UBC) Zone 2 provisions. Some of the structures and components were later evaluated for a safe shutdown earthquake (SSE) of 0.24 g. It is planned that the ATR remain in operation through the first decade of the next century, so an assessment of the additional risk posed by earthquakes is in order. The seismic accident sequence analysis performed for the ATR includes

Table 1. Comparison of ATR and PWR characteristics

Reactor Operating Conditions	ATR	PWR
Power (MW_{th})250	2000–4000	
Core power density (MW/l)	1	0.1
Operation pressure (psig)	355	2250
Inlet temperature (°F)	125	550
Outlet temperature (°F)	170	600
Primary coolant flow rate (gpm)	<48,000	300,000
Primary coolant weight (lb)	600,000	450,000
Primary coolant weight/thermal power (lb/MW)	2,400	170
Decay Heat (MW at 10 s)	13	135
(MW at 1 d)	1.3	19
Fuel		
Total Uranium weight (lb)	89	180,000
Enrichment (% U–235)	93	2–4
Configuration	48 in. long Al plates attached to side plates	Zirc rods containing stacked pellets
Matrix	UAl_x	UO_2
Fuel temperature (°F)	430	2000–3000
Fission product inventory at 250 MW ATR	60–d operation	10 times

a unique fault tree based treatment of seismically induced fires and floods.

METHODOLOGY

Through the following six steps, this complete seismic risk analysis considers all factors necessary to estimate the frequency of fuel damage at ATR:[1]

- Determine the earthquake hazard (hazard curves and site spectra)

- Identify accident scenarios for the ATR that lead to fuel damage (initiating events and event trees)

- Determine failure modes for the ATR safety and support systems (fault trees)

- Determine fragilities (probabilistic failure criteria) for the important structures and components

- Compute the frequency of fuel damage by developing and quantifying the accident sequence Boolean equations

- Estimate uncertainty in the fuel damage frequency.

HAZARD ANALYSIS

The INEL site covers approximately 2300 km^2 of southeastern Idaho. The ATR is located in the southwestern part of the INEL.

The INEL site lies in the eastern end of the Snake River Plain. This relatively aseismic area is bordered on the south, east and north by the Intermountain Seismic Belt (ISB), a region of general seismic activity that includes parts of five states and stretches from Arizona to Montana. The most recent earthquake in the ISB was the surface wave magnitude 7.3 Borah Peak event in October of 1983, which occurred in central Idaho about 120 km northwest of the Site and registered an acceleration of .025 g at the ATR.

ATR site specific hazard curves were not available. Therefore, the EBR–II site specific hazard curves developed by Risk Engineering, Inc. were used.[2] The EBR–II is located at the INEL Site approximately 25 km east of the ATR.

INITIATING EVENTS

The Level 1 internal events analysis for the ATR PRA, completed in December of 1989,* included development of accident scenarios that lead to fuel damage. For the seismic analysis, two unique event trees were developed to model damage to fuel in the core and in the storage canal. The core fuel damage event tree as shown in Figure 1, based upon the loss of commercial power event tree from the internal events analysis, includes explicit modeling of seismically induced loss of coolant accidents (LOCAs).

LOCA events in the primary system piping were divided, based on their effects on availability of safety systems. The safety system, such as the low pressure demineralized water system, provides the coolant to the core as well as to the fuel storage canal. Therefore, the state of the safety systems for the reactor was modeled explicitly in the canal event tree. The storage canal event tree was based on the loss of canal integrity event from the internal event analysis.

FAULT TREES

The system models created for the ATR PRA were modified using the guidelines provided in NUREG/CR–4826 for use in the seismic analysis.[3] First, all events that did not appear in the final dominant accident sequences of the internal events analysis, and whose unavailability was less

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* Unpublished research results of the Advanced Test Reactor Probabilistic Risk Assessment, EG&G Idaho, Inc., 1990.

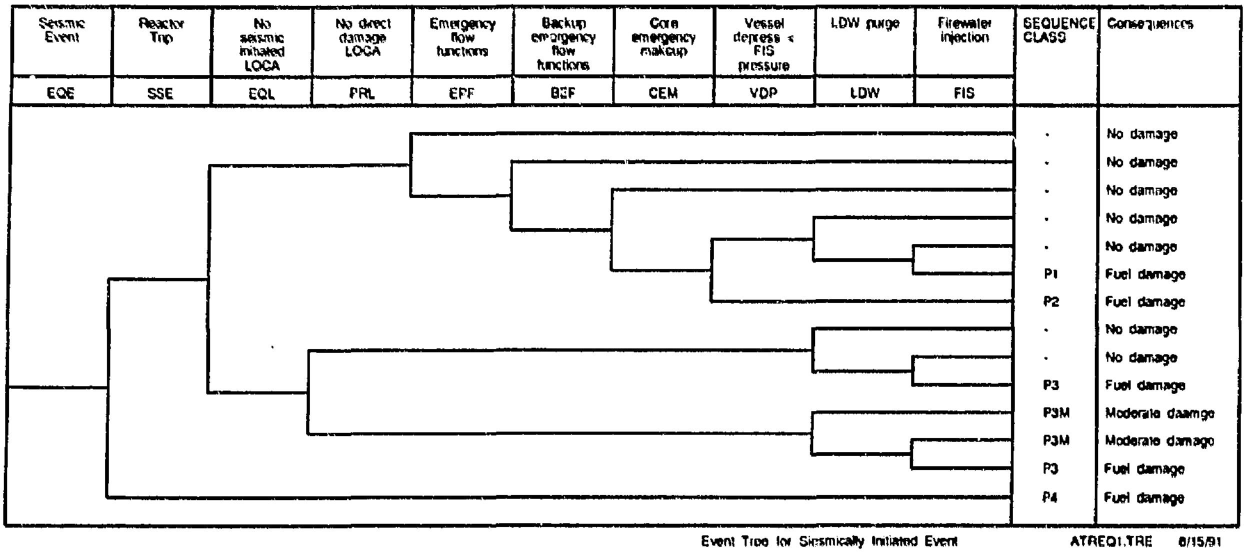

Figure 1. Seismic event tree.

than 1.0E–04, were eliminated from the fault tree models. Exceptions were made for common cause events.

Then, seismic failure modes were added for all major components and structures, for example, pumps, tanks, diesels, block walls, etc. The unique characteristic of the analysis was the explicit modeling of spatial effects of seismically induced internal flood and fire at the fault tree level, for example, loss of a safety pump due to seismically induced failure of the nearby fire protection piping. The seismic fragility of the fire protection piping, was explicitly modeled as one of the failure modes of a component, due to water intrusion or flooding. The propagation of the flood, was also modeled explicitly. Similarly, spatial effects of the seismically induced fire, and the effects of the actuation of the fire suppression system on nearby safety components, were modeled explicitly at the fault tree level.

FRAGILITY ANALYSIS

Seismic fragilities, for the important structures and equipment that were modeled in the event and fault trees, were developed by EQE. The steps involved in the fragility analysis were: (a) review of the plant design and seismic qualification information, (b) plant walkdowns, (c) a simplified seismic response analysis to obtain realistic floor response, and (d) estimation of the fragility parameters A_m, β_R, and β_U, of structures and equipment.

The review of plant design criteria provided the following information: The ATR buildings and the overhead storage tank were analyzed using Housner's spectrum anchored to an SSE of 0.24 g. No specific seismic criteria were used for the equipment within the buildings.

Two plant walkdowns were conducted for the ATR seismic PRA. The first walkdown encompassed a review of components identified by EG&G Idaho in the initial equipment list, along with the supporting structures. Only components that could be accessed were reviewed. The second walkdown included the review of the majority of components which could not be accessed during the first walkdown, and additional equipment items identified in the interim.

The procedure used in the walkdowns was to generally examine the structures for lateral resistance, look for seismic supports and bracing of equipment, visually confirm the adequacy of anchorage of equipment, and search for potential spatial seismic interactions. Physical dimensions of equipment were taken in most cases; these were supplemented by photographs of the equipment taken prior to and during the walkdown by EG&G Idaho and EQE. The walkdown revealed instances of unanchored equipment, unreinforced masonry walls in the vicinity of critical equipment, equipment supported on vibration isolators, and potential for seismic failures of fire water piping contributing to risks from seismically induced fires and floods.

Previous seismic analyses of ATR had assumed that the reactor building substructure located at or below grade is rigid, and that floor response spectra are the same as the ground spectra. Preliminary review performed in the present study indicated that the substructure fundamental horizontal frequencies are in the 10 to 20 Hz range. To account for ground motion amplification, a median centered response analysis was performed to obtain realistic floor spectra for equipment fragility evaluation. The median NUREG/CR–0098 ground response spectra for rock sites were used as seismic input in this response analysis.[4]

Over 90 seismic basic events, that is structures and equipment, were modeled in the seismic event and fault trees. However, based on the walkdown and comparison with median floor spectra to the ground motion spectra, approximately 45 seismic basic events, which were found to have median ground acceleration capacities greater than 2.0 g., were judged to not contribute significantly to seismic risk of ATR, and screened out from the quantification.

Seismic fragilities were developed for the remaining components by identifying credible failure modes and estimating the seismic margins arising from different sources, for example, strength, ductility, equipment response, and structural response. Both medians and variabilities were evaluated for these margins.

Finally, the median ground acceleration capacity of each component along with the values of logarithmic standard deviation of capacity β_R, such as randomness, and logarithmic standard deviation of uncertainty β_U in the median capacity, were derived. Knowing these parameter values, the complete family of seismic fragility curves for the component could be developed (see Figure 2). It was found that more than 15 components had median seismic capacities less than 0.3 pga. As a result of upgrading these components, the risk reduction was subsequently studied.

HUMAN RELIABILITY ANALYSIS

The human actions that occur after the earthquake were evaluated in a two step process. All the human actions were given a screening evaluation using the ASEP methodology.[5] Those actions that appeared to be important following the initial sequence analysis were given a more rigorous treatment using THERP[6] and HCR.* Twelve human actions which take place after the earthquake were evaluated in this manner. All twelve events have certain aspects in common. First, an earthquake of sufficient magnitude to fail commercial power is postulated. Second, both the running and the backup diesel generators fail, and the emergency backup diesel is called upon to start. (There are three diesel generators. Two diesel generators are assumed to fail at a moderate level of earthquake and are assumed to be unavailable from the beginning of the transient.) These twelve human actions were modeled assuming a

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* Unpublished research results on the Human Cognitive Reliability Model for PRA Analysis, NUS Corp., December 1984.

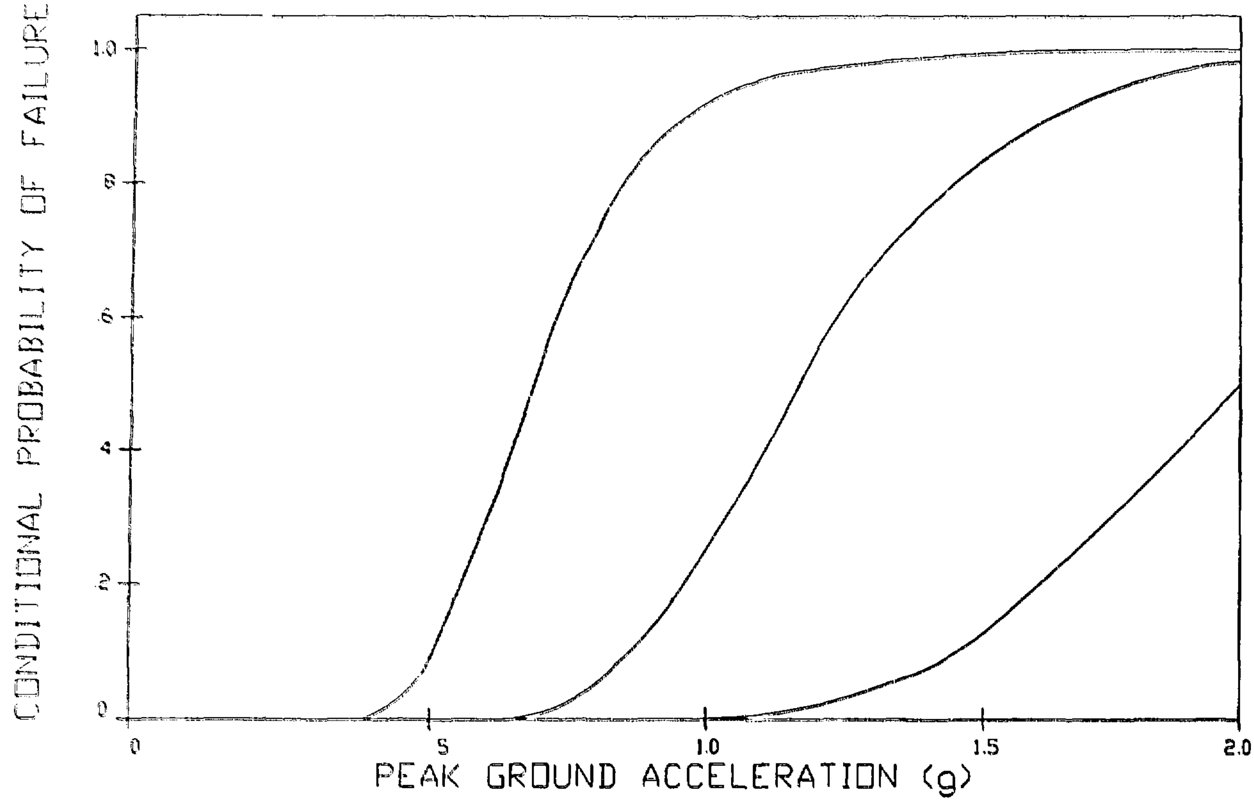

Figure 2. Fragility curve for a selected component (A_m =1.19 g, β_R = 0.22 and β_U = 0.31).

moderate stress level, as a result of an earthquake below the SSE value, and assuming a severe stress level as a result of an earthquake greater than the SSE value.

QUANTIFICATION

The CAFTA code package was used to develop accident scenario equations with initiators and all seismic failure mode events.[7] These equations were then quantified over the entire spectrum of possible earthquakes by EQE using a process that combined the sequence boolean logic, the seismic hazard curves, the seismic fragilities and nonseismic unavailabilities, including human actions. Uncertainty analysis using numerical integration (Discrete Probability Distribution) was performed.

RESULTS/INSIGHTS

The following insights were gained from the preliminary analysis:

Fuel damage frequency from the seismically induced fire and flood is less than 7% of the total fuel damage frequency estimated for the seismic analysis. A sensitivity analysis indicated that if penetrations through the diesel pit, which is located above the switchgear room are upgraded, then the contribution to the fuel damage frequency would be reduced to less than 2%.

A seismically induced failure of the surge tank was identified to result in a LOCA. Proper support to this tank would eliminate this LOCA sequence, and contribution to the fuel damage frequency would be reduced by 21%.

Human errors contributed 9% to the total fuel damage frequency from the seismic risk. However, when the analysis was carried out only for the severe stress level, the contribution to the fuel damage frequency increased by 24% and decreased by 8%, when the analysis was carried out only for the moderate stress level.

Total contribution from only the seismic failures (nonseismic failures were removed from the Boolean equations) was estimated to be 90% of the total mean core fuel damage frequency estimated from the seismic risk.

Low median capacity components, which could be fixed at relatively low cost, were removed from the Boolean equations. The analysis results indicated over 70% reduction in the final core fuel damage frequency from the seismic risk.

The contribution to the total fuel damage frequency (core and storage pool) from the canal event, was found to be negligible.

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[7] CAFTA Manual for Version 2.0, Science Applications International Corporation, February 22, 1989.