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On the Advisability of an Automatic Seismic Scram

W. J. O'Connell, J. E. Wells

Prepared for
U.S. Nuclear Regulatory Commission

Lawrence
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ABSTRACT

We examine the advisability of requiring commercial nuclear power plants to have a seismic scram system set at a high trip level--e.g., 0.6 SSE. A high trip level is intended to ameliorate one disadvantage, namely, loss of electric generation after an earthquake when a trip is not necessary for reactor protection. There are also advantages to a seismic scram. A seismic trip will give a lead time before other trip initiators. A few seconds is a significant time--3 s required to scram, and 5 to 10 s for 50% reduction of stored heat in the fuel rods. Then transient pressure and loads will be reduced. Using a decision tree, we compare the risks involved in both employing and not employing a seismic scram system. For a hypothetical site a seismic scram system significantly reduces the probability that an earthquake will cause a core melt, but it increases the probability that after the earthquake electric generation for community services and for operating the nuclear power plant's safety systems will be lost. Realistic assessments will require site-specific and design-specific data involving both power-generating and safety systems. We survey other countries' requirements on seismic scram and current U.S. regulations on seismic instrumentation.

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GLOSSARY OF ACRONYMS

ac	Alternating current
ACC	Accumulator(s)
AFW, AFWS	Auxiliary feedwater (system)
ANS	American Nuclear Society
ANSI	American National Standards Institute
BWR	Boiling water reactor
CANDU	Canadian deuterium and uranium reactor
CFR	Code of Federal Regulations
CM	Core melt (used in this report only)
CP	Charging pump
CVCS	Chemical and volume control system
dc	Direct current
DG	Diesel generator
FSAR	Final safety analysis report
GETR	General Electric Test Reactor
HPCI	High pressure coolant injection
HPIS	High pressure injection system
LOCA	Loss of coolant accident
LPIS	Low pressure injection system
LWR	Light water reactor
NRC	Nuclear Regulatory Commission
OBE	Operating basis earthquake
PORV	Power-operated relief valve
PWR	Pressurized water reactor
RHR, RHRS	Residual heat removal (system)
RPS	Reactor protection system
S/RV	Safety/relief valve
S/RV-O	O signifies failure to open
S/RV-R	R signifies failure to close
SNUPPS	Standardized nuclear unit power plant system
SSE	Safe shutdown earthquake
SSMRP	Seismic safety margins research program
SSR	Secondary steam relief
SW, SWS	Service water (system)

T1	Transient initiation with the power-conversion system available as a heat-removal path (an SSMRP term)
T2	Transient initiation with the power-conversion system unavailable (an SSMRP term)
WNP-2	WPPSS nuclear project No. 2
WPPSS	Washington Public Power Supply System

EXECUTIVE SUMMARY

We examine the advisability of requiring commercial nuclear power plants to have a seismic scram system set at a high trip level--e.g., 0.6 SSE. A high trip level is intended to reduce the frequency of one disadvantageous consequence of a seismic scram system, namely, loss of electric generation after a moderate earthquake when a trip is not necessary for reactor protection.

Earthquakes are of concern in the Eastern as well as Western U.S. In the East, earthquake occurrence is less frequent, but the area affected may be much larger.

Current U.S. regulations specify seismic instrumentation for timely information and evaluation. If the OBE (operating basis earthquake) design basis has been exceeded, the plant must be shut down for inspection; a normal shutdown procedure may be used.

In Japan nuclear power plants are required to have a high-level seismic scram system, set at 66% to 90% of the design basis earthquake. Many industrial facilities as well, such as railroads, city gas networks, and petrochemical plants, have seismic trigger systems for alarm and shutdown. In Italy seismic scram systems are not required; however, the operating utility has provided such systems for two nuclear power plants constructed in the 1960s. A seismic scram system is not planned for Italy's newest reactor at Caorso. Canada and the Federal Republic of Germany do not require seismic scram systems.

A seismic trip signal, if installed, would be one of several monitored parameters feeding into the existing reactor scram system. In a scram of a large power reactor, about 3 s is required for the control rods to be inserted and to shut down the chain reaction. The stored heat in the fuel rods and in the reactor then declines toward hot standby values, with time constants of about 5-10 s. Monitored parameters other than a seismic trip system could trip the reactor in the event of an earthquake, but trip levels of these parameters require specific frequency content and the buildup of vibrations. A high-level seismic trip (although not giving any lead time relative to strong motion initiation) would usually give lead time of 5-20 s before other trip initiators, such as turbine trip or loss of offsite ac power. This lead time is sufficient to achieve significant changes in reactor state--3 s to scram and 5-10 s for 50% reduction of stored heat.

To identify the possible advantages of a seismic scram system we must look at the possible transients and accident sequences that could lead to core melt and off-site exposure. The normal course of transients, including pressure and temperature behavior and relief valve operation, can be found in plant safety analysis reports. Possible accident sequences are the subject of several recent risk analyses. An early reactor trip that anticipates some other trip will reduce the transient pressure and loads and the core's stored heat. Consequently, fewer safety/relief valves (SRVs) will have to operate, and safety-related component failures (e.g., SRVs stuck open, or turbine-driven pumps unavailable) will be less probable. In case of a LOCA, an earlier trip will mean a cooler fuel rod temperature transient and a lower pressure; hence, less fluid will be lost in the blowdown phase before safety injection system operating pressure is reached.

There would be disadvantages to a seismic scram system. A seismic trip would be more likely to disable the offsite ac power, which may be needed for the reactor's safety systems as well as for offsite emergencies. In some cases a reactor trip and transient would be started when none would occur without the seismic scram system. For a multi-unit site or a wide-area earthquake, many electric sources could be tripped off at the same time.

A decision tree method is used to compare the risks of employing and not employing a seismic scram system. A realistic analysis requires site-specific and design-specific data. For a hypothetical plant, and using data and estimates from several sources, we find that a seismic scram system would reduce the probability of an earthquake-induced core melt accident by roughly a factor of three. This type of accident is significant, since an earthquake affects both the core and the containment protective systems at the same time. As for the disadvantages, the probability of loss of power generation in an earthquake emergency, and of simultaneous earthquake and scram loads, is increased. For the hypothetical plant analyzed here, that probability is about 2×10^{-4} per year. That disadvantage is somewhat offset by the cases (about 5×10^{-4} per year) where simultaneous earthquake and transient loads are reduced in magnitude by the seismic scram.

The value of a particular seismic scram installation requires a site- and design-specific evaluation. An evaluation for the purpose of choosing an optimum trip level would include data on the response of some non-nuclear-safety systems, such as the power conversion system and the regional power grid. Thermohydraulic calculations for a range of reactor trip times and LOCA sizes and locations would be useful to quantify the benefit of a seismic scram in these cases.

If a seismic scram system is planned, attention can be given to reducing the disadvantages, such as the chance of losing function of the offsite power network.

Attention in safety planning for an earthquake could be directed, not only to a seismic scram system, but to the security of power sources, both from offsite and onsite, and to planning the preferred mitigating and restoring procedures for a wide range of potential disturbances, rather than the limiting cases.

SECTION 1.0 INTRODUCTION

This is the final report of a study by the Lawrence Livermore National Laboratory on the advisability of requiring a high-level seismic trip (scram) system on commercial nuclear power plants. Previous studies[1-3] have addressed (1) the feasibility of providing a seismic trip system, and (2) the advisability of a low-level seismic trip system. This report addresses the advisability of seismic trip systems using a high-level set point (0.6-0.7 SSE). We studied several aspects of implementing a high-level seismic trip system:

- The likelihood that existing plant instrumentation will cause a trip during an earthquake.
- The timing of such trips relative to a seismic trip.
- Reactor responses and timing following a trip, especially for completion of scram, system actions, and pressure and temperature changes.
- The desirability of allowing nuclear power plants to continue to generate power during an earthquake.
- The possibility of spurious reactor trips caused by a seismic trip system.

Using a decision tree, we compared the risks involved in both employing and not employing a seismic trip system.

In Section 2, background information is provided to give you a better understanding of the problems encountered from earthquakes. In Section 3, we discuss the current and proposed standards and guides dealing with seismic instrumentation and seismic data evaluation. In Section 4, we discuss the current policies and plans of several foreign countries with respect to the seismic scram question. In Section 5, we discuss some existing seismic scram systems in the United States. Section 6 provides a list of the advantages and disadvantages of having an operational seismic trip system. Section 7 contains a discussion of the decision analysis concerning a seismic trip system installation. The qualitative list of advantages and disadvantages in Section 6 and existing fault tree analysis are used to determine the outcomes and probability values on the decision tree. Section 8, the final section, provides a list of results and implications.

SECTION 2.0 BACKGROUND

2.1 INTRODUCTION

Large earthquakes can cause serious reactor safety problems. It was the purpose of this study to determine if reactors could be made safer by installing a system (automatic seismic scram system) that would scram the reactor when a predefined acceleration level was reached. In order to make such a determination, many facets of this problem have to be studied. For example, if the seismic trip system could detect a high-level earthquake and then scram the reactor, the shutdown might reduce the risk of an accident. However, it is also possible that when the peak strong earthquake motion arrives the combined seismic and shutdown stresses will actually cause an accident to occur. Another point is that as the intensity of earthquake increases, so does the probability that the plant will trip on some other variable (a transient or a variation in a monitored parameter in the instrumentation system).

2.2 EARTHQUAKE CHARACTERISTICS

Earthquakes are a familiar part of the "California experience." In the Eastern and Central U.S.[4] they are less frequent but still a hazard. Comparing the Western U.S. with the Eastern and Central U.S. we note:

1. The largest recorded earthquakes in both areas are of comparable magnitude.
2. The frequency of earthquakes of a specified magnitude is much larger in the Western U.S.
3. The attenuation with distance from the epicenter is much less in the Central U.S.
4. For a specified site, the frequency of recurrence of an earthquake with a specified peak acceleration is higher in the Western U.S. (The frequency of recurrence of low and moderate acceleration values is better established. The frequency of the rarer high accelerations is estimated indirectly and is more uncertain.)

An example of acceleration-frequency relationships for the Zion, IL site[5] is shown in Fig. 2.1. There is an uncertainty band of a factor of two in acceleration above and below the best-estimate curve shown.

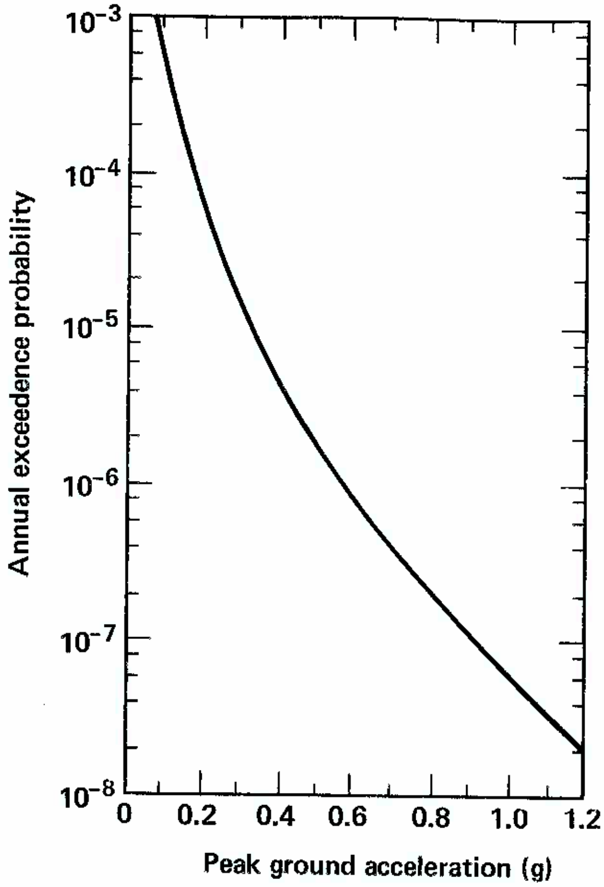


Figure 2.1. Best estimate of SSMRP seismic hazard curve for the Zion site.

The acceleration value for the design SSE is usually lower in the Eastern and Central U.S. than in the West. Hence, the frequency of reaching 1/2 SSE is not consistently different in the East and West. The frequency of reaching 1/2 SSE is, for many plants, roughly between 10^{-4} per year and 3×10^{-3} per year. Both site-specific differences and uncertainty in estimation contribute to this range of values. Of the events greater than 0.5 SSE, about 80% to 90% will be below 0.9 SSE. On the other hand, there are about ten times as many earthquakes in the range 0.15-0.5 SSE than earthquakes greater than 0.5 SSE.

2.3 SEISMIC RESISTANCE

The safety-related systems of a nuclear power plant, i.e., those necessary for safe shutdown, prevention of accidents, etc., are designed for the SSE load with very high design and verification standards. The probability of failure of any of these systems at an SSE intensity is very low. Their probability of failure increases for those exceedingly rare earthquakes well above the SSE level.

The non-safety-related systems (e.g., the power generation systems and the utility-wide power distribution grid) are not designed to the same earthquakes by the same standards. These systems may be designed to the standards of the Uniform Building Code.

2.4 SEISMIC RESPONSE

A low-level earthquake might cause no disturbance at all. A slightly larger earthquake might trip the power generation system and hence the reactor. The trip might be caused by turbine vibration, an off-site power grid transient, or the tripping or malfunction of some component in the power generation system or the plant auxiliaries. A higher intensity earthquake might damage some non-safety-related equipment. An even higher intensity earthquake, at or above the SSE, might damage a few safety-related components, but diverse redundant components could maintain the safety-related functions. A very high-intensity earthquake might damage enough components to cause an accident; such low-probability events are analyzed in studies such as the Seismic Safety Margins Research Program (SSMRP).[5] These effects of an earthquake are not exactly correlated with peak acceleration; earthquakes differ both in their characteristics and in the responses of structures to them.

2.5 SEISMIC TRIP

The main purpose of a high-level seismic trip (say at 0.6 SSE) would be to anticipate events that might lead to an accident, getting the control rods in and reducing the stored heat of the core at an early time. The system would also anticipate transients in the power-conversion system. For example, in some cases scramming the reactor first would lead to a milder transient.

There are also some disadvantages to a seismic trip. The advantages and disadvantages will be discussed in Section 6.

2.6 TIMING OF INITIATING EVENTS

The most likely initiators of a reactor trip and a transient (in the absence of a seismic scram system) are

1. Turbine vibration trip (if vibration monitor is set to trip the turbine automatically).
2. Disturbance in ac power to plant and systems (a transient in the offsite grid, damage in the plant switchyard for offsite ac power, or damage to the in-plant distribution system).
3. Disturbance in some component of the power conversion system, e.g., some mechanical or electrical failure.

The most likely components to fail will probably not do so immediately on occurrence of a ground acceleration; failure usually requires the buildup of component vibrations to a high value or repeated cycles for fatigue or cumulative damage. This process requires a duration of, say, 5 to 20 seconds. For further discussion see Section 5.4.

2.7 TIMING OF A SCRAM

The time from the moment a parameter reaches its trip level to when the rods are inserted to 85% of their full insertion (enough to shut down the chain reaction) ranges from 2.7 to 4.2 s for a large power reactor, depending on the initiating parameter. (These are conservative estimates of the timing for licensing use.) For a representative small reactor, GE's Vallecitos material irradiation reactor, the corresponding scram time is 0.5 s.

After control rod insertion in a large power reactor the heat in the fuel rods decreases with about a 10 s time constant.

The timing in a scram is described further in Section 5.2.

2.8 TURBINE TRIP

Turbine thrust bearings are sensitive to lateral forces usually arising from unbalanced turbine vibration.[6] Vibration monitors either trip the turbine or notify the operator. The trip levels and timing are related to turbine system integrity rather than to earthquakes. There have been numerous instances, however, when an earthquake tripped the turbine vibration monitor. For example, in the northern Kentucky earthquake[7] of July 27, 1980, the

J. M. Stewart fossil power plant which sustained some superficial damage to chimneys had all four units tripped due to suspected turbine vibration. Another fossil plant nearby had no damage and no turbine trip. In the Imperial Valley earthquake of October 15, 1979, at the El Centro oil-powered generating plant[8] a lightning arrestor fell across some on-site AC power lines, shorting them and tripping the boilers and turbine-generators before turbine vibrations had an effect.

In nuclear power plants a turbine trip or a step decrease in external load on the generator does not always require a reactor trip. It depends on the power level.

In the SNUPPS power plant (Ref. 9, Sec. 1.2.8) the turbine bypass (steam dump to the condenser) is sized to handle 40% of full power. If there is a step decrease in external load of as much as 50% of full power, this can be accommodated (40% in steam dump, 10% decrease in reactor power) without tripping the turbine or reactor. In the WNP-2 BWR power plant, (Ref. 10, Sec. 1.2.2.6) the turbine bypass is sized to handle 25% of full power; then a loss of external load or a turbine trip at 25% power or less will not require a reactor trip. The turbine stop valve (and main steam isolation valve) positions are connected into the reactor protection system to trip the reactor when they start to close; this trip signal may be bypassed when the reactor is at a low power within the capacity of the turbine bypass. In any case, a rise in steam pressure beyond its limit value will trip the reactor.

2.9 NON-SEISMIC SCRAMS

A study of LWR scram experience[11] indicates a record of ~2.5 scrams per year from above 20% of full power, after an initial "working in" or learning period (see Fig. 2.2.).

SECTION 3.0 CURRENT STANDARDS AND GUIDES

3.1 INTRODUCTION

The Code of Federal Regulations (CFR)[12] Section 10CFR100 Appendix A defines the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE), how they are to be determined, and how they are to be applied to

3.2 OPERATING BASIS EARTHQUAKE

The OBE is "that earthquake which considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant." Its function in design is "that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional."[12]

This wording does not imply that the power generation features must remain operable, but rather that if the plant continues to operate, the safety-related systems necessary for assuring the health and safety of the public shall remain functional.

The power generation equipment beyond the main steam isolation valves in a PWR (Ref. 9, Sec. 3.2) or beyond the isolation valves and turbine stop valves in a BWR (Ref. 10, Sec. 3.2) is generally considered not to be seismic category 1 (although piping up to the next anchor is included in the seismic analysis). The turbine bypass line, for example, in both the PWR and BWR is considered to be in NRC Quality Group D and is designed to the ANSI B31.1 Power Piping Code.[13] This is the code applicable to fossil-fueled power plants. This code does include consideration of earthquake load "where applicable." The earthquake load if any is to be specified by the owner in the piping Design Specification, is not a nuclear safety question and is not discussed in the plant Safety Analysis Report. The WNP-2 FSAR (Ref. 10, Sec. 3.2.1) does state that for non-seismic category I structures, systems and components not analyzed for the SSE and OBE, the seismic loading conditions are determined from the Uniform Building Code and used in their design where applicable.

3.3 OBE EVENT DETERMINATION AND SHUTDOWN

The requirements in the CFR for shutdown and inspection are developed in more detail in the standard ANSI/ANS-2.10-1979[14] and in draft regulatory guide EM-706-5.[15]

If an earthquake occurs, plant operation may continue if the earthquake level is less than the OBE and the plant is seen to be still capable of operating normally and within the Technical Specifications. If the plant has

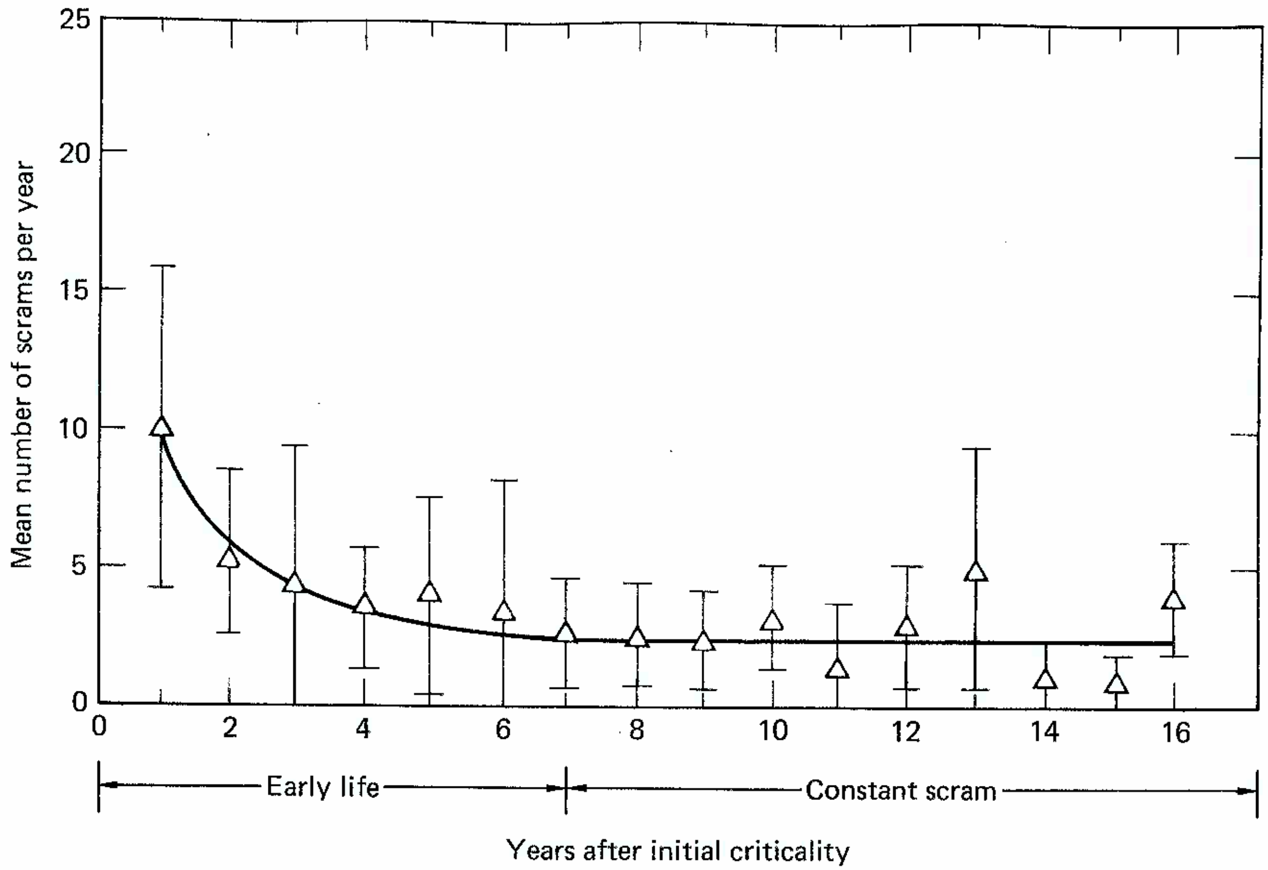


Figure 2.2. Annual number of reactor scrams in LWRs from above 20% of full power.

engineering design. This section of the CFR specifies requirements for shutdown if there is an earthquake larger than the OBE, and for demonstrating the absence of functional damage to relevant features before resuming operations. This section also requires suitable seismic instrumentation so that the response of plant features important to safety can be determined promptly for comparison with the plant design bases, for decision on continued operation, and for such timely action as may be appropriate. ANSI/ANS standards and NRC regulatory guides develop more detail on earthquake instrumentation criteria and on the processing and evaluation of records obtained from seismic instrumentation to determine whether an earthquake exceeding the OBE has occurred and to evaluate responses in comparison to the original OBE seismic design bases.

not tripped during the earthquake, then the immediate evaluation is based on the accessible seismic data. If the OBE level is exceeded in any of this data, then the plant must be shut down for inspection; a slow normal shutdown procedure may be used. If the OBE level is not exceeded, other inaccessible seismic data can be retrieved at the next outage.

3.4 SEISMIC INSTRUMENTATION

The Standard ANSI/ANS-2.2-1978,[16] NRC Regulatory Guide 1.12,[17] and Standard Review Plan[18] Section 3.7.4 provide guidance on the seismic instrumentation required for timely information, evaluation and decision making after an earthquake. These guides cover instruments which transmit information to the control room and recording instruments whose records can be retrieved when access is possible. The intent is to provide enough information for timely action if necessary and for evaluation of whether the OBE level has been exceeded. Instrumentation for a seismic scram is explicitly not prescribed but is reserved for future consideration.

SECTION 4.0 REQUIREMENTS OF FOREIGN COUNTRIES

A high-level seismic scram system is required for nuclear power plants in Japan. In Italy seismic scram systems are not required; however, the operating utility has provided such systems for two nuclear power plants constructed in the 1960s. A seismic scram system is not planned for Italy's newest reactor at Caorso. Canada[19] and the Federal Republic of Germany were queried, and they do not require seismic scram systems.

4.1 JAPAN'S SEISMIC SCRAM SYSTEMS

In Japan many industrial facilities, such as railroads, city gas supply networks, and petrochemical plants, have seismic trigger systems for alarm and shutdown.[20] Nuclear power plants have seismic triggers for shutdown at a level of 66% to 90% of the design basis earthquake. Usually two-out-of-three coincidence of sensors is required.[20, 21]

4.2 ITALY'S SEISMIC SCRAM SYSTEMS

Two nuclear power plants in Italy, Latina and Garigliano, have low-level seismic scram systems.[22] They trip at 0.03 g and require two out of three coincidence. At Garigliano the earlier sensors of about 1970 vintage sometimes tripped from earthquakes below the desired 0.03-g level because the sensors were oversensitive at frequencies near 1 Hz. A new system installed in 1980 operated during the earthquake of November 26, 1980, but has had no spurious trips to date.

5.0. SEISMIC SCRAM SYSTEMS DESCRIPTION

An automatic seismic scram system combines two parts: a seismic trigger and a reactor scram system. The scram system acts to insert the control rods rapidly and shut down the chain reaction when any monitored parameters go beyond specified threshold levels.

In this Section we first describe the Diablo Canyon Plant seismic scram system. Then we describe reactor scram systems--monitored parameters, time sequence, possible trip initiators in the event of an earthquake, and the timing relations between a seismic trip signal and the other possible trip initiators or accident-initiating events.

5.1 DIABLO CANYON SEISMIC SCRAM SYSTEM

The Diablo Canyon PWR has a seismic scram system.[23] There are three triaxial seismic acceleration detectors at diverse locations near and in the reactor building. If any two of the three detectors signal an acceleration above the action level (0.35 g in the free field, which is 47% of the SSE level), then the reactor scram system is activated. The scram of the reactor then trips the turbine generator, and the turbine bypass valves open. The reactor decay heat is removed through the steam generator, with the steam in the secondary circuit bypassing the turbine and going into the condenser.

5.2 REACTOR SCRAM SYSTEM

A reactor scram system is designed to shut down the reactor's chain reaction rapidly and reliably whenever key parameters of the reactor system go beyond specified threshold levels. Tables 5.1-5.3 list typical monitored parameters for a PWR, BWR, and CANDU[24] reactor system, respectively. The

Table 5.1. Typical trip signals--PWR reactor.

1.	High neutron flux.
2.	High rate of power increase.
3.	High coolant temperature.
4.	High or low system pressure.
5.	High power.
6.	Low coolant flow.
7.	Loss of coolant.
8.	Low steam generator water level.
9.	Low steam generator pressure.
10.	Safety injection initiation.
11.	Turbine trip (except once through steam generator design).
12.	Loss of plant auxiliary power.
13.	Manual trip.

Table 5.2. Typical trip signals--BWR reactor.

1.	High reactor containment pressure.
2.	High reactor pressure.
3.	Low water level in the reactor vessel.
4.	High neutron flux.
5.	Rapid closure of turbine control valves.
6.	Closure of the turbine stop valve.
7.	Main steam isolation valve closure.
8.	High radiation in the steam line.
9.	Loss of plant auxiliary power.
10.	Manual trip.

Table 5.3. Typical trip signals--CANDU reactor.

1.	High neutron power.
2.	High rate log neutron power.
3.	High heat transport pressure.
4.	Low heat transport pressure.
5.	High building pressure.
6.	Low steam generator level.
7.	Low pressurizer level.
8.	Low gross coolant flow.
9.	Low boiler feed line pressure.
10.	Manual trip.

parameter-monitoring, activation-logic, and reactivity-insertion mechanisms are designed with redundancy and diversity for reliable operation when needed. For many parameters, a two-out-of-four monitor threshold level is required to initiate a scram; this criterion provides sufficient redundancy while guarding against a spurious scram, and it allows channel testing during reactor operation.

In a PWR, the fast insertion of negative reactivity is done by a fast release of the drive rod and a drop of the control rod/drive rod unit under gravity.

According to the SNUPPS reactor Final Safety Analysis Report (FSAR) (Ref. 9, Sec. 15.0), the timing of a PWR scram is as follows. The parameter activates a signal, releasing the control rods, which drop into the reactor core. When the rods have dropped 85% of their length, they have inserted 50% of their negative reactivity, enough to cause the shutdown of the chain reaction. The maximum time for each step leading to rod insertion is shown in Table 5.4. After the chain reaction is stopped, the heat output in the primary coolant circuit declines with a time constant of about 10 s (see Fig. 5.1). Immediately after shutdown, the decay heat power is 7% of the preceding reactor power and decreases with time[25] (see Fig. 5.2).

In a BWR, the control rod unit (Ref. 26, Sec. 4.2) consists of a neutron-absorbing control rod and a drive rod which extends down to the control rod drive housing mounted below the reactor pressure vessel. Upon a

Table 5.4. Time delays--PWR scram.

1.	From signal reaches trip level to rod trip (rods released)		
	(a)	On neutron flux	0.5 s
	(b)	On turbine trip	2.0 s
2.	From rods tripped to rods inserted		
	(a)	Rods 85% inserted	2.2 s
	(b)	Rods 100% inserted	2.9 s

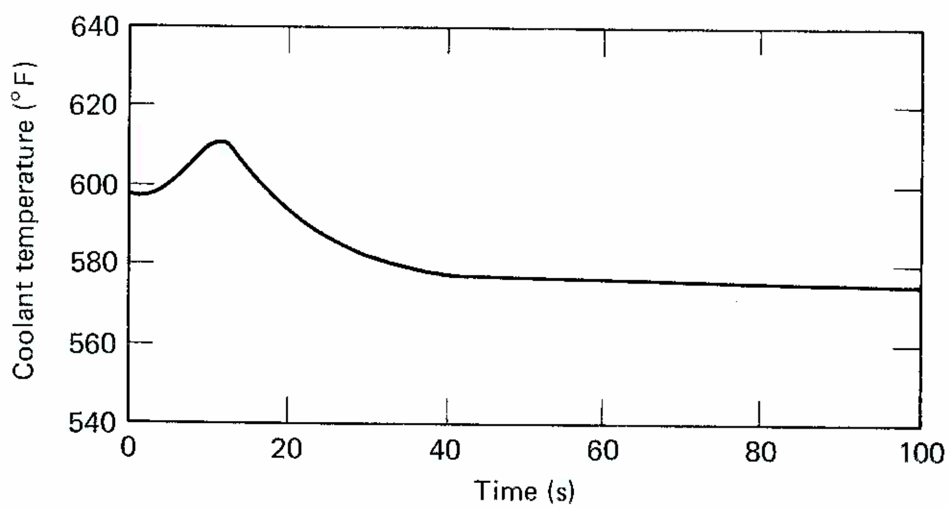


Figure 5.1. Decline of PWR coolant average temperature after a reactor trip. For the transient shown a reactor trip on a pressure signal follows 11 s after the initial turbine trip at 0 s.

scram signal, a high-pressure hydraulic system pushes the control rods up in their channels between the fuel rod bundles. Water in the control rod drive housing displaced by the drive piston's motion is let out to a special reservoir. The timing for a BWR scram (Ref. 10, Chapter 15) is shown in Table 5.5. Figure 5.3 shows the timing of negative reactivity insertion. As in the PWR case, the heat output from the fuel rods declines with a time constant of about 10 s (see Fig. 5.4).

We will not discuss various alternatives for the insertion of the control rods and other negative reactivity, in PWRs and BWRs, for the unlikely case where the first scram system might malfunction. Neither will we discuss incidental moderating phenomena; for example, in a BWR, stopping the jet recirculation pumps and feedwater pumps will stop the sweeping of steam

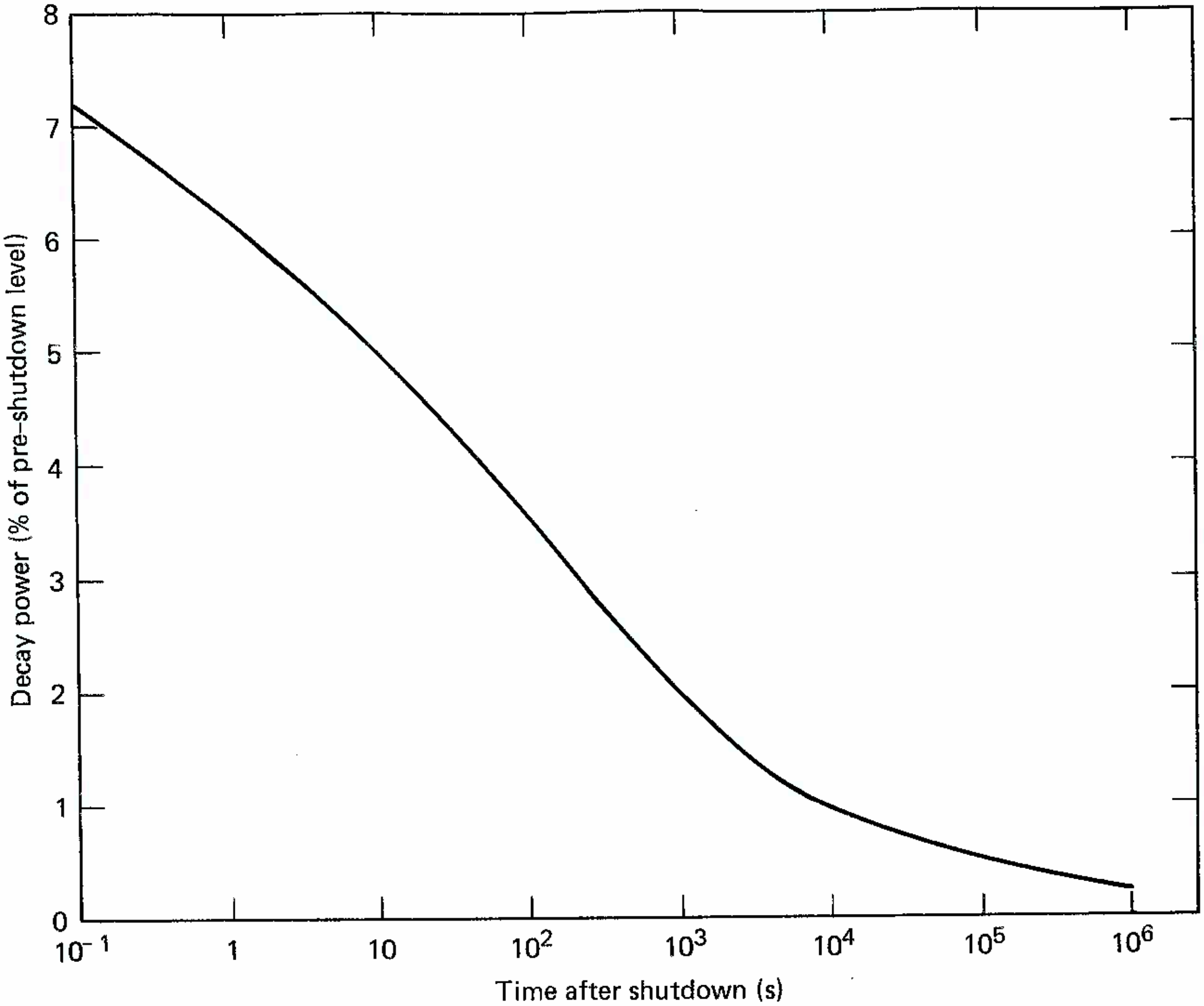


Figure 5.2. Thermal power generation from radioactive decay after reactor shutdown.

Table 5.5. Time delays--BWR scram.

1.	From signal reaches trip level to control rod scram system actuated	
	(a) On neutron flux	0.11 s
	(b) On turbine trip	0.08 s
2.	From rods tripped to rods inserted	
	(a) Rods 85% inserted	3.4 s
	(b) Rods 100% inserted	4.0 s

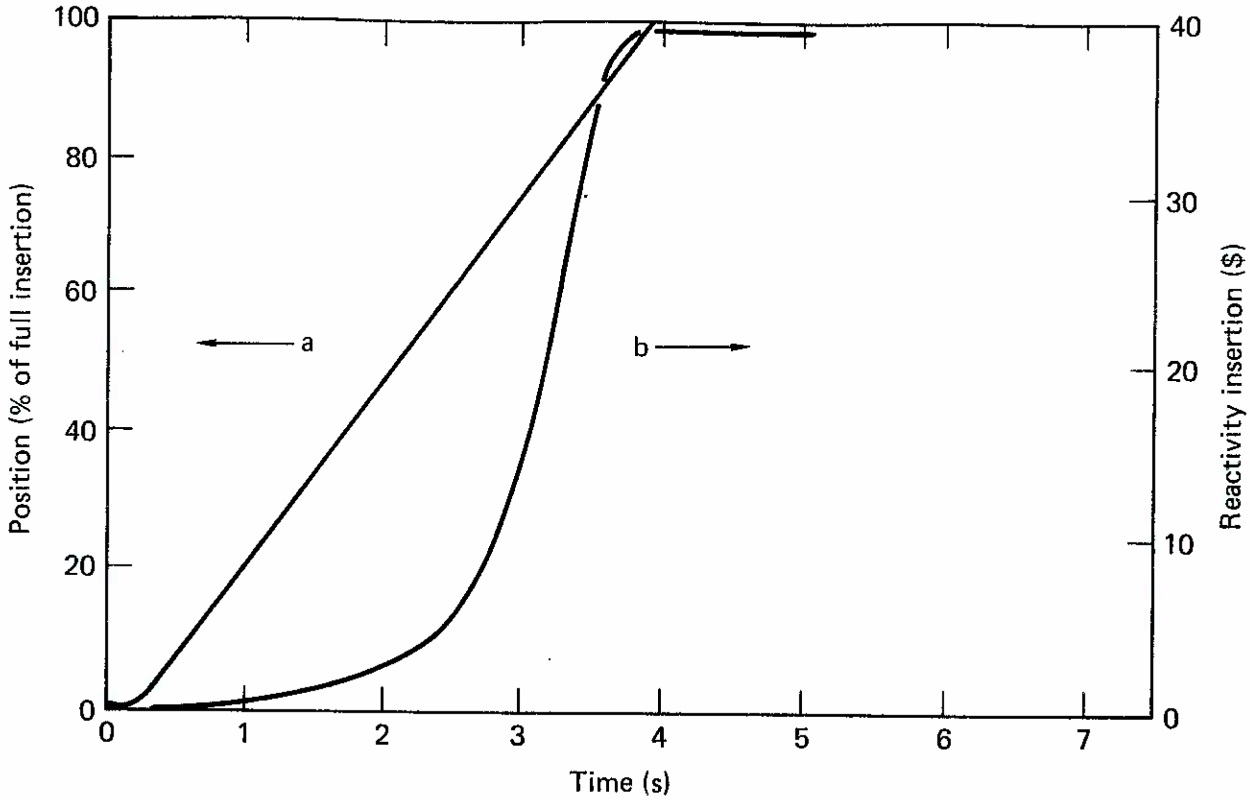


Figure 5.3. Control rod position (a) and reactivity insertion (b) vs time for a BWR scram.

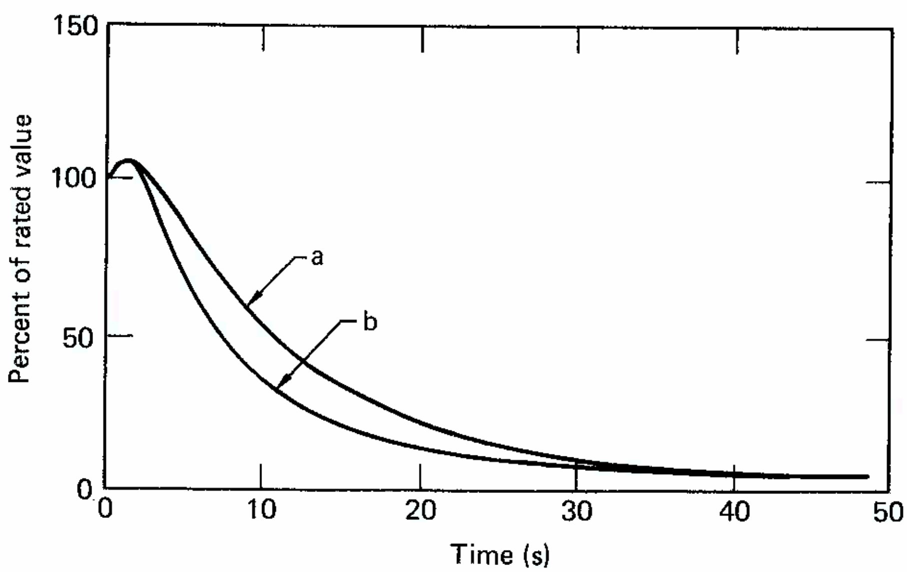


Figure 5.4. Decline of BWR fuel temperature after a reactor trip. The transient shown has a turbine trip at 0 s and a reactor scram initiated at 0.1 s. Curve (a) is peak fuel center temperature. Curve (b) is average surface heat flux.

bubbles out of the core. If the turbine is still drawing steam, the pumps' stopping can reduce core power from full power to below 40% of full power (Ref. 10, Fig. 15.2-7).

In a PWR, a reactor trip will initiate a turbine trip; conversely, a turbine trip first will cause a reactor trip if the reactor is near full power. In the WNP-2 BWR, a reactor trip does not immediately trip the turbine. The turbine is tripped when the turbine inlet steam pressure drops below a threshold.

In a CANDU reactor, the control rods are above the reactor core. Upon a trip, the rods are dropped into the core. Seismic tests in Japan[27] on a one-fifth scale model of a CANDU reactor showed rod insertion times scaled to a full size CANDU as follows:

- With no earthquake, 1.5 s.
- During a 1-g earthquake shaking (using the 1940 El Centro record scaled up to 1 g), 1.6 s, compared to a design specification of ≤ 2 s.

5.3 GETR SEISMIC SCRAM SYSTEM

The GE Test Reactor (GETR)[28] at the Vallecitos Nuclear Center near Pleasanton, California, is a small reactor which has a seismic scram system. The design criteria of the reactor and its seismic scram system differ from those of the Diablo Canyon power reactor.

The GETR reactor is a materials-irradiation reactor with a power of 50 MW (th). Its core is 2 feet in diameter and 3 feet in height, compared to a height of 18 feet for a large power reactor. It operates at 180°F and 150 psig pressure. The reactor vessel is braced within a larger pool of water. Valves separate the core volume from the larger pool of water. Upon a scram or other shutdown, these valves open. Cooling then takes place via natural circulation.

The seismic trigger signal is based on two multi-axial acceleration detectors, with a trigger level of 0.01 g, one of the detectors being sufficient to trigger a scram. The control rods drop 3 feet into the core, 2 feet being enough to shut down the reactor. The timing of a scram is shown in Table 5.6. The total time is 0.5 s, a few seconds less than for a PWR scram.

The present seismic triggers at GETR have experienced several low-level earthquakes, including four since October 1977. In all cases the seismic trigger functioned reliably.[28]

Table 5.6. Time delays--GETR scram.

1.	From signal (seismic) reaches trip level to rod trip.	0.18 s
2.	From rods tripped to rods inserted	
	(a) Rods inserted 2 feet	0.3 s
	(b) Rods inserted 3 feet	0.5 s

5.4 OTHER TRIP INITIATORS AND TIMING IN THE EVENT OF AN EARTHQUAKE

The most likely initiators of a reactor trip and a transient (in the absence of a seismic scram system) are judged to be

1. Turbine vibration trip (if vibration monitor is set to trip the turbine automatically).
2. Disturbance in ac power to the plant and its systems (a transient in the offsite grid, damage in the plant switchyard for offsite ac power, or damage to the in-plant distribution system).
3. Disturbance in some component of the power conversion system, e.g., some mechanical or electrical failure.

The most likely components to fail will probably not do so immediately on occurrence of a ground acceleration; failure usually requires the buildup of component vibrations or repeated cycles for fatigue or cumulative damage. This process requires an earthquake lasting for, say, 5 to 20 s. Fig. 5.5, for example, shows one component of the El Centro 1940 ground acceleration and of a piping valve's resulting motion with fundamental frequency 2.9 Hz and 2% damping. Buildup requires 3 s. At 22 s even stronger vibrations occur, from buildup of the structural vibrations moving the pipe supports, and then buildup of the piping vibrations.

The components most likely to initiate a trip include

1. Turbine vibration. The turbine-vibration monitors detect lateral vibration (one horizontal direction and vertical) of the shaft relative to its alignment along the bearings' axis. Even with a low vibrational trip level, buildup of turbine lateral modal vibration might take 2-10 s. The monitor may either trip the turbine or notify the operator. At operation below 50% power, a turbine trip does not necessarily initiate a reactor trip.

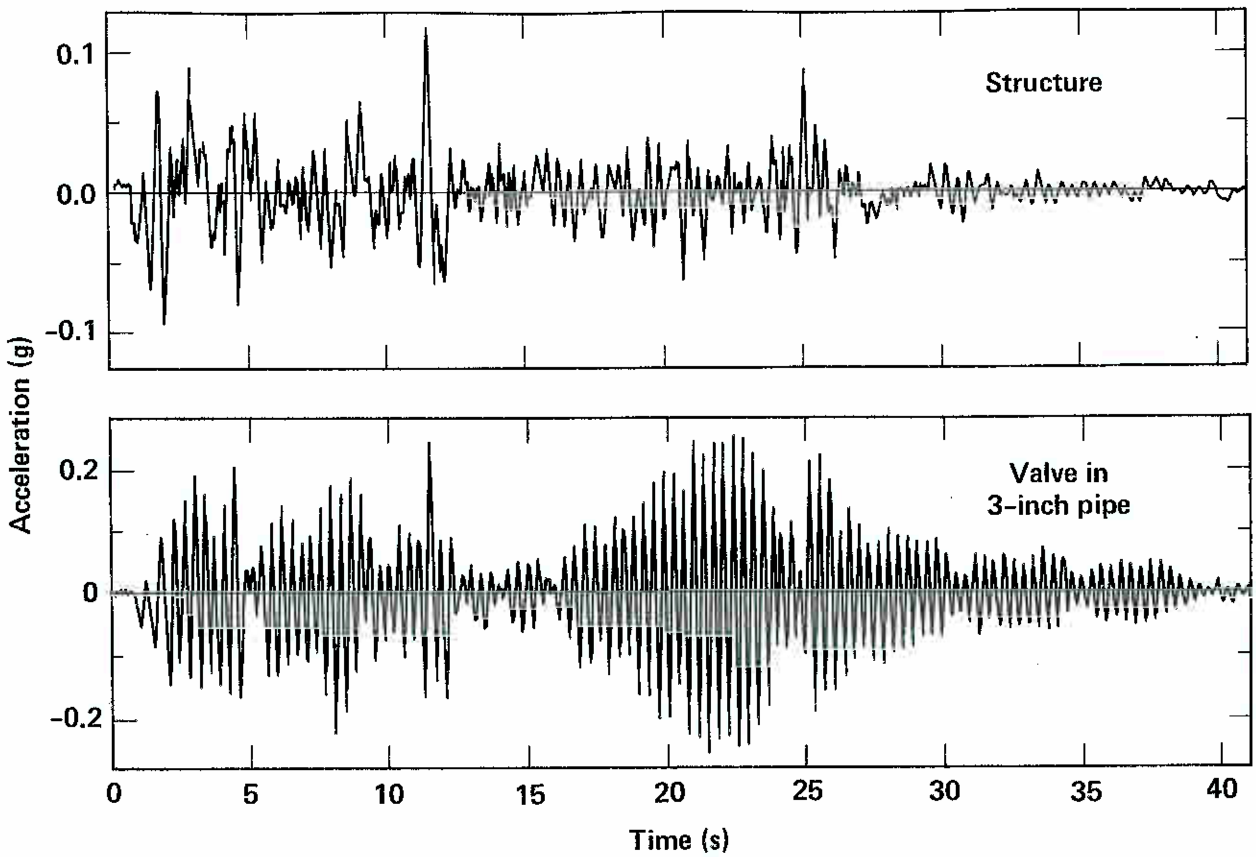


Figure 5.5. Example Zion time histories calculated from a ground motion based on the El Centro, CA, 1940 earthquake record.

2. High-voltage components in the ac power switchyard, mounted on stacks of ceramic insulators. These insulators could topple, causing short circuits or pulling cables loose. Conceivably, a single ground pulse, if large and if not reversed quickly enough, could topple them, but a more likely failure mode is through a buildup of vibration at the fundamental frequency of the mounted system.
3. In two recent earthquakes a lightning arrestor broke loose and fell across on-site power distribution lines. (El Centro power plant,[8] October 15, 1979 and Lawrence Livermore Laboratory substation[29], January 25, 1980). This type of failure requires time for the lightning arrestor vibrations to build up to a breaking point, plus a few seconds to fall.
4. Components in the power conversion system, e.g., auxiliaries such as motor control panels, cooling water. In the response spectra used for earthquake protective design, the peak spectral accelerations at

some frequencies are many times the peak ground acceleration. These higher component response accelerations cause more stress but require time to reach a peak, say 5 to 20 s. Note also that many component failures do not immediately trip the turbine or reactor. For example, loss of feedwater heating in a BWR (in manual control mode) leads to an increase in core neutron power. The core power gradually rises and eventually exceeds an upper trip level; the RPS trips the reactor. The time to trip is 60 s. (Ref. 10, Sec. 15.1.1) A specific example of initiation of a trip by involvement of the cooling system occurred with a German research reactor during an earthquake of Richter magnitude 5.5 on September 3, 1978. A relay trip cut off one main coolant pump; loss of the pump led to a primary circuit pressure drop which then led to a reactor scram.[30]

5. Offsite ac power supply grid. Swaying of towers or cables may cause cables to contact (short circuit) or break loose. Damage to distant substations may cause short circuits. Again, these are vibrational processes which require time to build up.

In summary, the initiating events most likely to occur, such as loss of the power conversion system, or loss of offsite ac power, would likely require times of 5-20 s after the start of an earthquake's strong motion. These time estimates are for a strong earthquake, say 0.7 SSE to 1.5 SSE. For a stronger earthquake of longer duration, a comparable component failure might occur sooner, say in 2-10 s. Other components which would not fail in the lesser earthquake might now fail, in times of 5-40 s, causing functional problems insofar as failing components might be redundant equipment supporting a safety function.

A turbine vibration trip might or might not occur before a more serious transient initiator such as loss of offsite ac power. The trip level is probably sensitive to earthquakes, but it responds only to one horizontal and one vertical direction, and mainly to earthquake frequency content near the turbine vibrational frequencies. Further, the turbine vibration trip is not a nuclear safety-grade system.

A high-level seismic trip would give a reactor trip lead time, relative to the transient initiators mentioned above, of possibly 5-20 s. The advantages of lead times will be discussed in Section 6.

5.5 SUMMARY

A seismic trip signal, if installed, would be one of several monitored parameters feeding into the reactor scram system. In a scram of a large power reactor, about 3 s is required for the control rods to be inserted and to shut down the chain reaction. The stored heat in the fuel rods and in the reactor then declines towards hot standby values, with time constants of about 5-10 s. Monitored parameters other than a seismic trip system could trip the reactor in the event of an earthquake, but these require specific frequency content and the buildup of vibrations. A high-level seismic trip (although not giving any lead time relative to strong motion initiation) would usually give 5-20 s lead time before other trip initiators, such as turbine trip or loss of offsite ac power. This lead time is sufficient to achieve significant changes in reactor state--3 s to scram and 5-10 s for 50% reduction of stored heat.

SECTION 6.0 ADVANTAGES AND DISADVANTAGES OF A SEISMIC SCRAM SYSTEM

This section identifies many of the advantages and disadvantages of a seismic scram system.

Broadly put, it would be preferable to know at the start of an earthquake whether during its course it will cause a reactor trip or an accident sequence initiation. If the earthquake is going to trip the reactor in any event, then an early seismic trip provides definite advantages, and only a few disadvantages. If the earthquake is not going to trip the reactor by initiating some other accident sequence, then the seismic trip itself is not useful, and has several disadvantages. This section lists the advantages and disadvantages; how to draw a balance on net advantage is described in Sec. 7.

6.1 EVENT SEQUENCES IN AN EARTHQUAKE

To identify the possible benefits, we must look at the possible transients and accident sequences which could lead to core melt and off-site exposure. The SSMRP has examined an example plant, the Zion PWR in Illinois, for system response to an earthquake.[31] (See Sec. 7 for further development.) Following a large earthquake, there would most likely be a transient, and possibly a loss-of-coolant accident (LOCA) (more likely a small rather than a large pipe break). Most of these initiating events will not develop into a core melt.

Among those transient sequences leading to core melt, the most likely sequence is one where the AFW/SSR system fails in its core-cooling function. This failure could be caused by loss of ac power to the AFW pumps and failure of the turbine-driven AFW pump system (e.g., by failure of a valve). This sequence would lead to a small slow loss of coolant fluid through the pressurizer SRVs, taking one to four hours before the top of the core becomes uncovered. Thus, in looking for advantages of a seismic scram, we look for ways in which the seismic scram would lower the probability of, delay, or reduce the consequences of this accident sequence.

In a LOCA in a PWR, the blowdown removes both fluid and heat energy, reducing the temperature and pressure. As the pressure is reduced, more injection systems can come into action, making up the fluid loss. Thus the LOCA usually does not lead to a core melt. In the SSMRP study, those LOCAs most likely to lead to core melt had a failure either of the emergency coolant-injection function or emergency coolant-recirculation function. In the small-small LOCA (pipe break diameter 0.5-1.5 in.) the blowdown is not large enough to depressurize the core. Then the AFW/SSR system is also needed; its failure could lead to core melt. This scenario is close to that of the transient leading to core melt.

Parallel to the reactor's accident sequence, there is a possibility of containment failure. In the earthquake event these two failure sequences are correlated, notably through the possible loss of all ac power, which would affect the transient and core cooling sequence and also the containment heat removal. (This dual sequence was also noted in another risk study of the Zion plant.[32]) As an alternative to the electric pumps Zion has a turbine pump for AFW.

In looking at advantages of a seismic scram preceding a LOCA, we look for reduced probability of safety system failure or reduction in severity of the consequences of the accident sequences.

6.2 ADVANTAGES OF A SEISMIC SCRAM

1. In the early seconds of a transient, a prior seismic scram will mean a lower core heat content and a lower pressure increase (or none) and hence:
   a. Fewer safety/relief valves operating.
   b. Lower combined transient and earthquake loads.
   c. Lower probability of safety system failures (e.g., SRVs stuck open, or turbine pumps unavailable).
2. In the early seconds of a LOCA, a prior seismic scram will mean lower fuel rod temperatures, lower fuel rod surface heat flux, lower core heat content, and lower pressure and hence:
   a. Less fluid lost in the blowdown phase.
   b. Lower fuel rod temperature transient. (The fuel rods stay covered with water in the early phase of a LOCA, so this temperature transient is affected only by the reduced water flow velocity and the initial fuel rod temperature.)
   c. In the (rare) case where the fluid flow must reverse (dependent on break size and location relative to emergency coolant injection location), the lower fuel and fluid temperatures will provide less thermal-hydraulic opposing force to the flow reversal.

6.3 ADVANTAGES OF A SEISMIC SCRAM--SUPPORTING INFORMATION

6.3.1 During the Early Time Stages of a Transient

A reactor trip prior to a turbine trip or other trip initiator will produce a milder transient in pressure and temperature. This will be illustrated by comparing different transients described in the FSAR, first for a BWR and then for a PWR.

For a BWR the transients in the FSAR (Ref. 10, Chapter 15) do not include a simple reactor trip; the closest comparable trip is one initiated by loss of feedwater heater (see Fig. 6.1). In this scenario the loss of the feedwater heating leads to a gradual rise in core power, causing a reactor trip in about 60 s. The turbine and the recirculation pump are not tripped at this time. The core power and the vessel pressure drop, with no momentary increase and no relief valve action required. The turbine control valves move toward closed

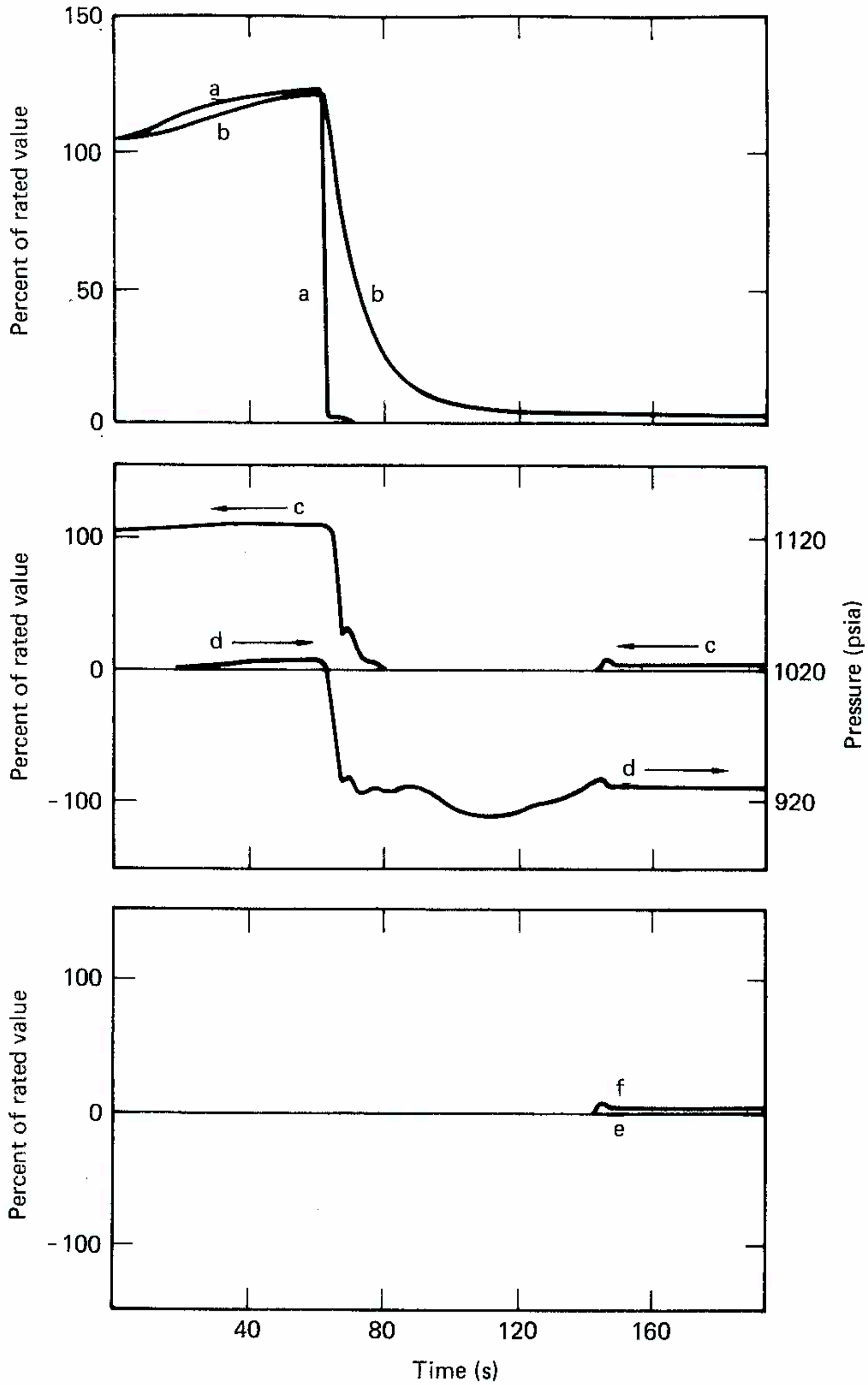


Figure 6.1. BWR transient after loss of feedwater heater with manual flow control. (a) core neutron flux. (b) peak fuel center temperature. (c) vessel steam flow. (d) vessel pressure (psi). (e) relief valve flow. (f) turbine bypass valve flow. The ordinate of all curves except (d) is percentage of rated full-power value; the abscissa is time after transient intitiation.

to maintain steam pressure, but the turbine does not trip. Thirty seconds later the turbine, recirculation pump, and feedwater turbines trip, causing no pressure rise.

A BWR turbine trip initiates trips of the reactor control rods, the recirculation pumps, and the feedwater turbines. In contrast to the trip described above, however (see Fig. 6.2), there is a vessel pressure rise and a momentary neutron power increase before the scram is effected. Even with the turbine bypass valves opening rapidly to accept 25% of full power, there is a vessel pressure rise of over 100 psi, requiring the opening of five relief valves. Reactor steam blows down into the pressure suppression pool in the primary containment, at almost 100% of full power for 3-5 s. The turbine bypass continues removing steam at 25% of full power for 30 s after the reactor scram.

The BWR loss-of-generator-load transient and loss-of-all-grid-connections transient (generator load and power to auxiliaries) are similar in effect to the preceding turbine trip.

Both the BWR turbine trip with turbine bypass unavailable and the transient from closure of all main steam line isolation valves are similar to the above described turbine trip transient in neutron power and fuel temperature. All the reactor heat removal, however, is now through the relief valves to the primary containment. The relief valves cycle open and closed. The containment pressure rises but not to its design limit.

In the PWR (Ref. 9, Chapter 15) transients follow a course similar to that for the BWR, except that there is no brief increase of neutron power. As a substitute for a simple reactor trip, we examine a transient due to a malfunction of a feedwater control valve (see Fig. 6.3). This malfunction leads to a gradual rise in core power, which eventually trips the reactor and thence trips the turbine. There is only a very small increase in pressurizer pressure (10 psi), followed by a drop, and a drop in core temperature.

In the PWR turbine-trip-initiated transients, the turbine bypass can take 40% of full power. The turbine trip initiates a reactor trip, which requires 2.7-4.2 s to become effective, depending on the actuation time and the rod drop time. The SNUPPS FSAR does not analyze all turbine trip scenarios, but only a bounding case which assumes (a) turbine bypass unavailable, and (b) reactor trip delayed until a pressure sensor initiates a trip. The first 3 s of the transient, however, will be the same for a prompt reactor trip. In their analysis (see Fig. 6.4), within these first 3 s the pressures in both

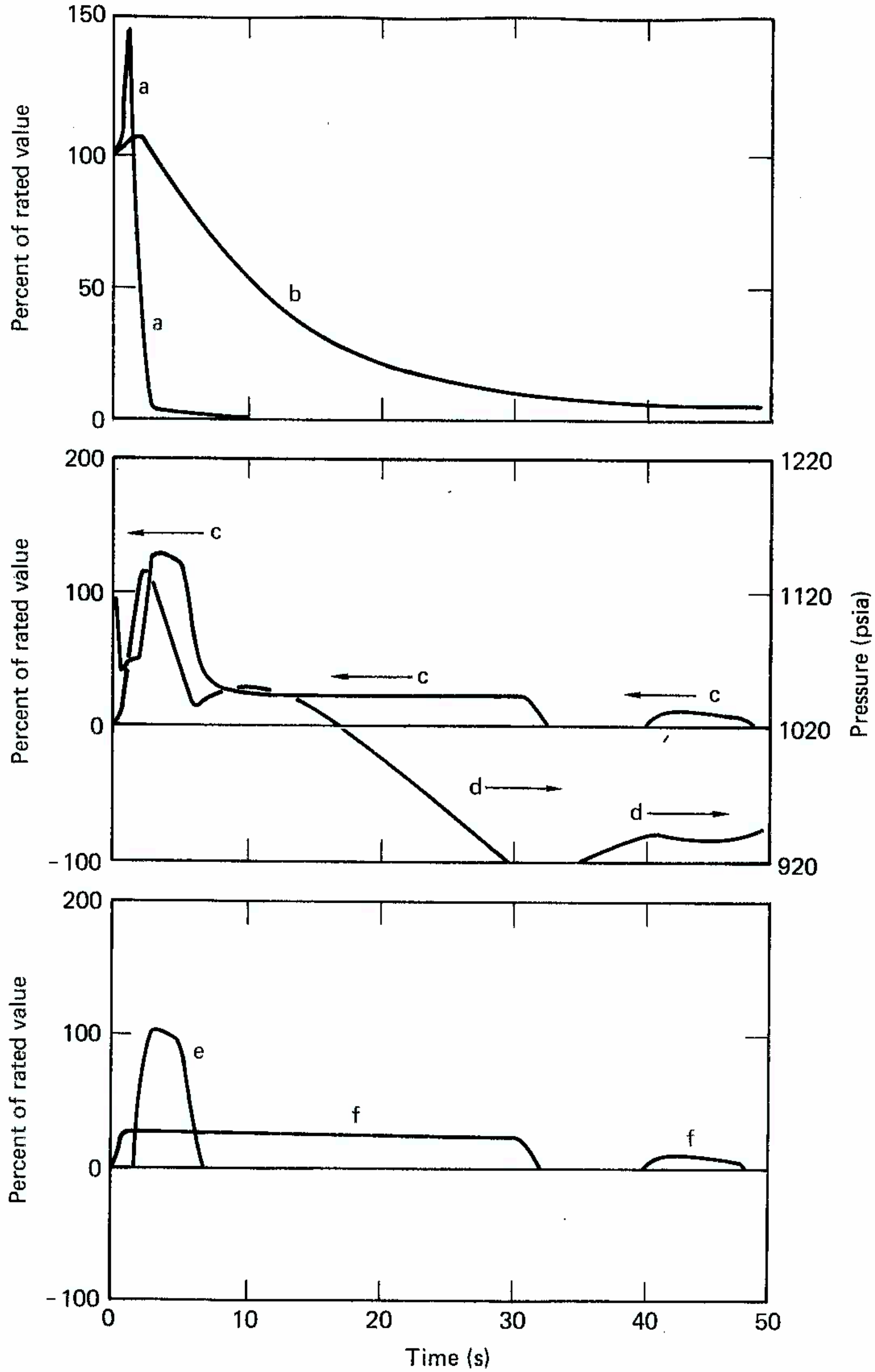


Figure 6.2. BWR transient on turbine trip and trip scram, with bypass on and recirculation pump trip. Curves (a)-(f) show the same variables as in Fig. 6.1.

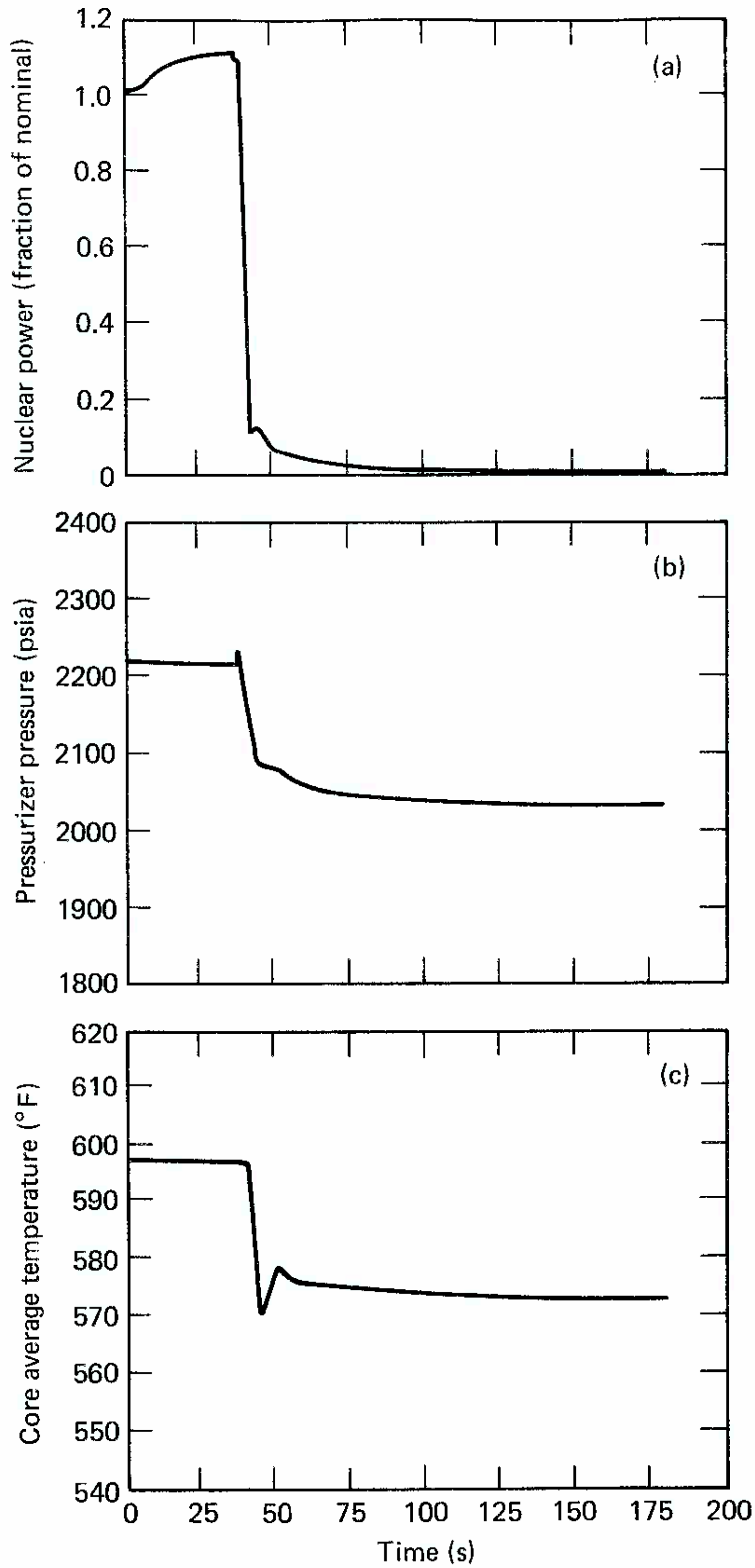


Figure 6.3. PWR transient on feedwater control valve malfunction, with later reactor and turbine trips. (a) nuclear power generation. (b) pressurizer pressure. (c) core average temperature.

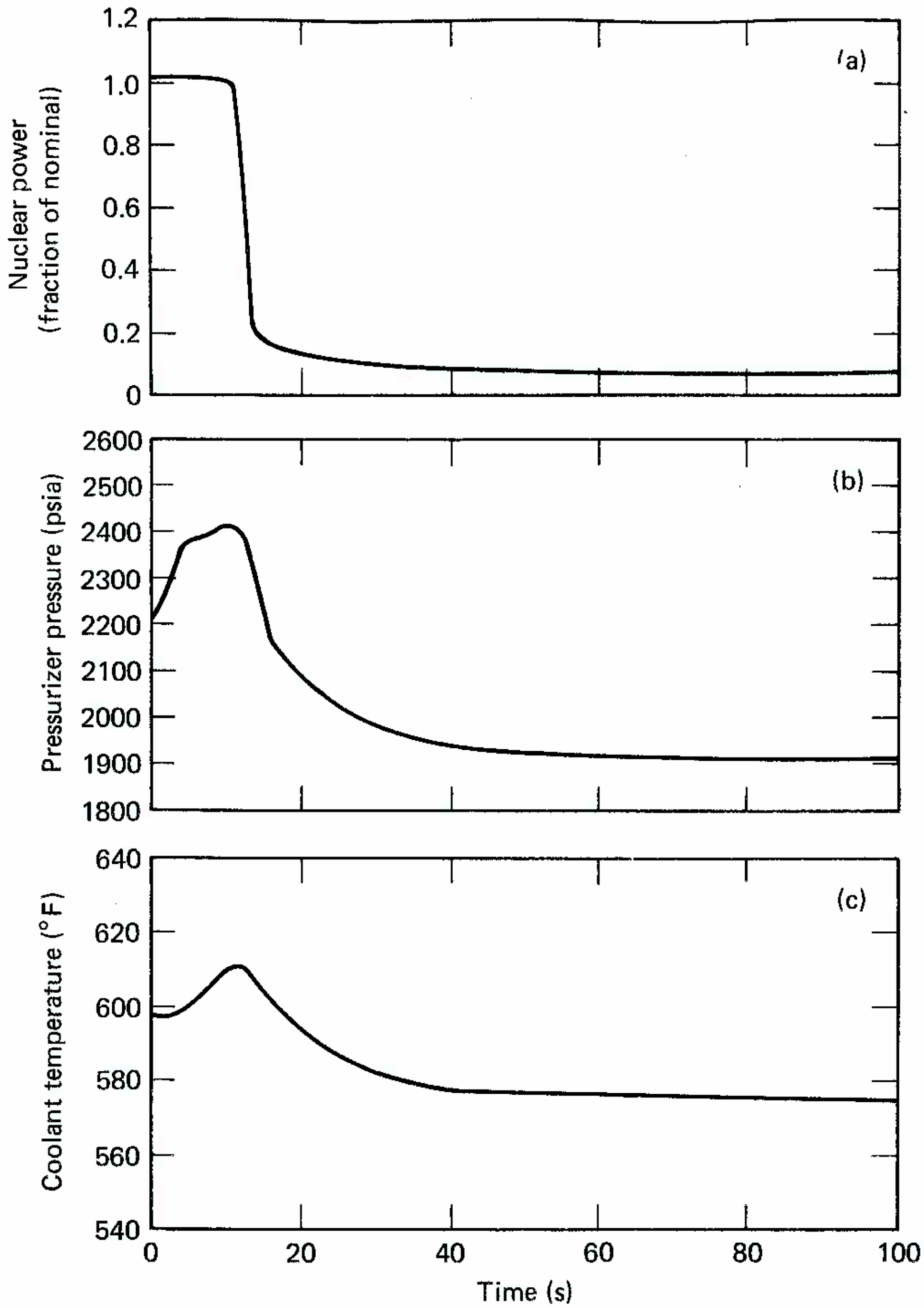


Figure 6.4. PWR transient on turbine trip, with later reactor trip from pressure sensor, with bypass off, pressurizer spray and PORV available, and with minimum reactivity feedback. (a) nuclear power generation. (b) pressurizer pressure. (c) core average temperature.

secondary and primary rise (primary by 180 psi). The steam generator PORVs or safety valves are required, opening to the atmospheric steam dump, and on the primary side the pressurizer spray and the pressurizer PORVs to the containment are required.

6.3.2 During the Early Stages of a LOCA

A LOCA will cause a reactor scram on reactor low pressure or on containment high pressure or temperature, with the timing depending on the size of the LOCA.

As noted in Section 5, a scram lead time of just 5 to 10 s will produce a great reduction in stored heat, as well as shutting down the heat generation from 100% to 7% and less. As shown in Fig. 6.3, a PWR scram will also reduce the primary system pressure and temperature. The pressure drops by 180 psi, which is 8% of the total pressure or 25% of the decrease required for operation of the high-pressure injection system (HPIS) (see Fig. 6.5). The core average temperature drops by 25°F, which is 22% of the decrease required for steam-water equilibrium at the operating pressure of the accumulators.

Thus the time from LOCA initiation to initiation of the HPIS or LPIS is reduced, reducing core coolant loss and diminishing the chance of uncovering the top of the fuel rods. At any time, the pressure is lower than it would be without the early scram, reducing the fluid blowdown rate and increasing the capacities of the injection systems. Hence, fewer injection systems might be needed for success. Clarification of this question would benefit from detailed thermal-fluid calculations.

For a small LOCA, a turbine trip before any other trip could exacerbate the situation by abruptly removing a cooling path and causing a few seconds of pressure rise, as noted in Section 6.3.1, above.

6.3.3 During the Later Stages of an Accident Sequence

A scram shuts off the fission heat generation and allows a reduction in stored heat. In terms of integrated heat energy, lead time of 10 s is worth 1000 s in the later stages of an accident (see Fig. 5.2).

A major contributor to the total risk of core meltdown, both from seismic and random initiators, is a set of conditions where both primary and auxiliary

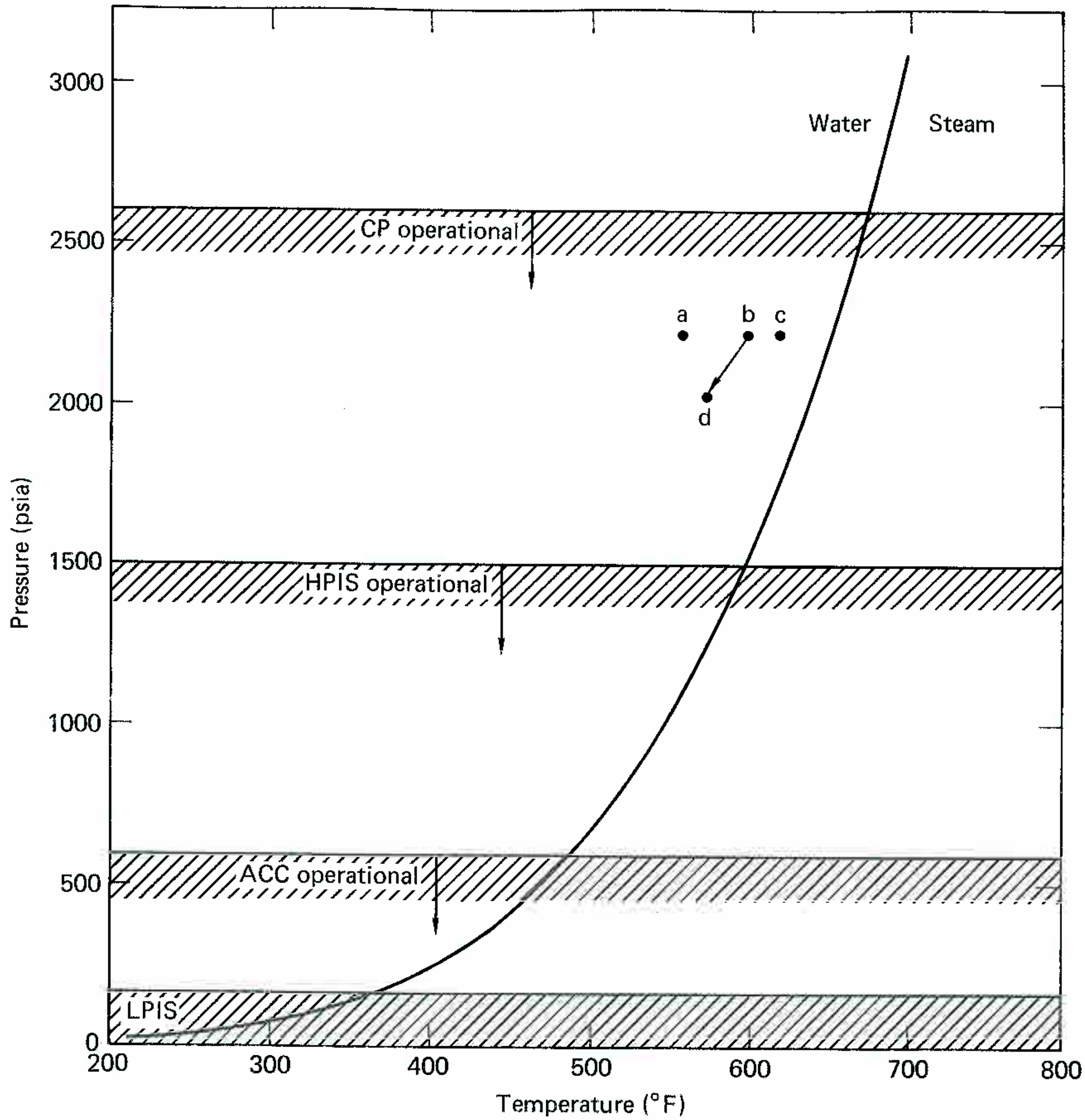


Figure 6.5. Water liquid-steam equilibrium curve. Points a, b, and c are PWR full-power values in cold leg, core average, and hot leg, respectively. Point d indicates core average values after reactor trip transient of Fig. 6.3. Shaded areas and below are operational ranges of CP, HPIS, accumulators, and LPIS.

cooling circuits are unavailable, and the reactor water boils away through relief valves, eventually exposing the top of the fuel.

A major random initiator is failure of all ac power sources. In a PWR study[33] failure of ac power and failure of the turbine-driven auxiliary feedwater pump leads to boiling dry of the steam generators, followed by a pressure rise in the primary system, so that the pressurizer PORVs or safety valves open. Then high-pressure blowdown is the only path of heat removal from the core. The high-pressure injection system is unavailable because of the loss of ac power. The reactor water boils away, exposing the top of the fuel at 45-100 min if no recovery is achieved. As noted above, an anticipatory scram 10-15 s before the scram on loss of offsite ac power would add 11-14 min in the later stage for recovery efforts.

A study of loss of all ac power for a BWR4 shared similar results.[34] In the most likely among the scenarios leading to core melt, the turbine-driven high-pressure coolant injection (HPCI) works, and heat is removed via blowdown to the containment. The HPCI would fail in 2-4 h, either because of dependence on the station dc batteries or because of the increasing pressure in the reactor. Then boiloff would proceed. Fuel exposure would occur in 300-350 min (as compared to 100 min in the PWR case). As in the PWR case, an anticipatory scram 10-15 s early would add 15-18 min to this time for recovery efforts.

The SSMRP study[31] for seismic risk in a PWR found a similar set of dominant scenarios for core melt risk. For earthquakes in the range 0.9-1.8 SSE, Fig. 6.6 shows the most likely of the scenarios leading to core melt. A transient that leaves the power conversion system unavailable as a heat sink successfully trips the reactor, but then the auxiliary feedwater and secondary steam relief system are also unavailable. The pressurizer safety or relief valves open for pressure limitation and as a heat-removal path, allowing some blowdown to the containment. This continues as a high-pressure blowdown with insufficient high-pressure makeup water, leading to core exposure and core melting. This scenario found for the seismic case ends in the same way as the scenario described above for a PWR with random failures.

In the above scenarios there are qualitative advantages of an anticipatory scram (seismic scram):

1. At early stages there is a milder transient and there are fewer relief valve operations.
2. At later stages there is a slower development toward a crisis stage, giving 11-18 min longer for recovery attempts.
3. In case of a LOCA, pressure and heat content are lower, leading to lower fluid loss and to earlier depressurization to a level sufficient for operation of emergency coolant injection systems.

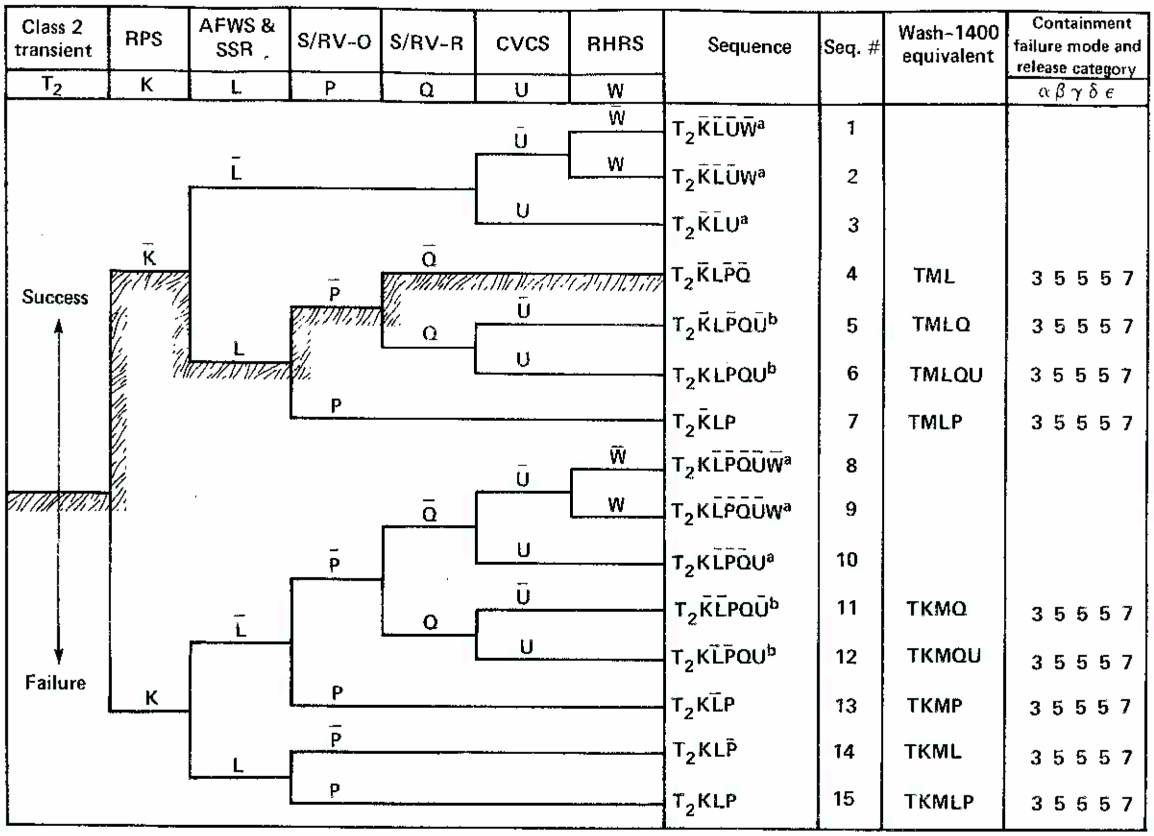


Figure 6.6. Class 2 transient event tree. The most likely event sequence among those leading to a core melt is shaded. (a) indicates sequences with no core melt. (b) indicates sequences leading into a small-small LOCA sequence.

6.4 DISADVANTAGES OF A SEISMIC SCRAM

The disadvantages of a seismic scram are

1. Electric generation sources would be lost more frequently than would be necessary (if we waited for a disturbance-initiated trip).
2. Coincident loads--earthquake and trip transient--would occur more frequently than otherwise.
3. There would be additional demands on safety systems. (A trip demands certain safety functions.)
4. For a multi-unit site or a wide-area earthquake, we would lose many electric sources at about the same time. This disruption might cause a power network or load imbalance leading to the loss of the network. The loss of network might affect the scrammed reactor and distant reactors. The higher the trip level the less severe this effect would be.

5. Decision on scram (shutdown) would be taken away from the operator.
6. There would be cost disadvantages:
   a) Initial and maintenance costs.
   b) Costs of reactor down time in case the seismic scram system (a safety system) is unavailable.
   c) Costs of possible (but unlikely) spurious scrams.
7. Having one more safety system to test before startup and to monitor might spread operator attention and training over a greater number of safety systems, possibly affecting non-seismic safety.

Disadvantages 1 through 4 apply to those earthquakes where no trip signal would be generated in the absence of the seismic trip system.

Disadvantage 4 applies also to earthquakes where other trip initiators will occur. In a multi-unit site with a seismic scram system all units will trip off the network at almost exactly the same instant. The network may not be able to accept such a transient. If more distant nuclear power plants on the network also have a seismic scram, they may trip off, even if they are marginally below the seismic level where they would trip anyway. The large loss of generating power may cause load imbalance and network failure. Thus the first reactor, which would trip in any case, would be more likely to suffer from the loss of off-site ac power needed for safety systems.

6.5 SUMMARY

An early reactor trip that anticipates some other trip initiator will reduce the transient loads. To be sure, the design allows for the larger transient loads (in coincidence with SSE, where some inelastic deformation and fatigue usage is allowed). However, with the smaller transient loads:

1. The small probabilities of component failure will be made even smaller.
2. Combined earthquake and transient loads will be smaller.
3. Inelastic deformation will be less frequent, allowing an earlier plant restart.

Then there will be a smaller probability of failure sequences (leading to core melt and radiological release). Also the heat content will be lower, giving:

1. Less severity to some steps in the sequence.

2. In case of LOCA, lower pressure and less fluid loss in blowdown.

3. More time for recovery measures.

There would be disadvantages. Disabling the off-site ac power would be probable, and in some cases a reactor trip and transient would be started when none would occur without the seismic scram system.

A balance of the advantages and disadvantages will be described in Section 7.

SECTION 7.0 DECISION ANALYSIS

In this section we outline a method which should assist in deciding whether or not to install a seismic scram system at a nuclear power plant. The method draws together the advantages, disadvantages, and the likelihood of alternative future events. We carry out a sample calculation and evaluation. The numerical values are site-specific; they are drawn from several sources to illustrate the method.

7.1 DECISION ANALYSIS METHOD

The method[35] involves building a decision tree--identifying decision forks, where one of several alternative actions can be chosen, and chance forks, where alternative future events branch out. We need to (1) know the probabilities of the alternative future events, given the preceding decisions and events, (2) the value or utility of the outcome of each future event sequence, and (3) any initial costs of chosen actions. We then can compare the value of the different choices in terms of their outcome probabilities and utilities. In some applications the expected utility of a choice is calculated in an actuarial approach. In other applications several categories of utility are retained without trying to weigh them on a common unit such as cost. Categories of utility may include economic cost, radiation exposure, and low consequence/high probability events vs high consequence/low probability events.

7.2 DECISION TREE

The decision tree for installing a seismic scram system is presented in Fig. 7.1. There is only one decision fork, the first fork in the tree, since we are considering here only that measure and not additional steps which may be worth considering in earthquake planning (see Section 8.2). The following forks are chance forks, representing the alternative events in each step of a possible future event sequence:

1. Earthquake.
2. Seismic trip or other trip.
3. Other transient or LOCA initiator.
4. Successful conclusion or core melt.

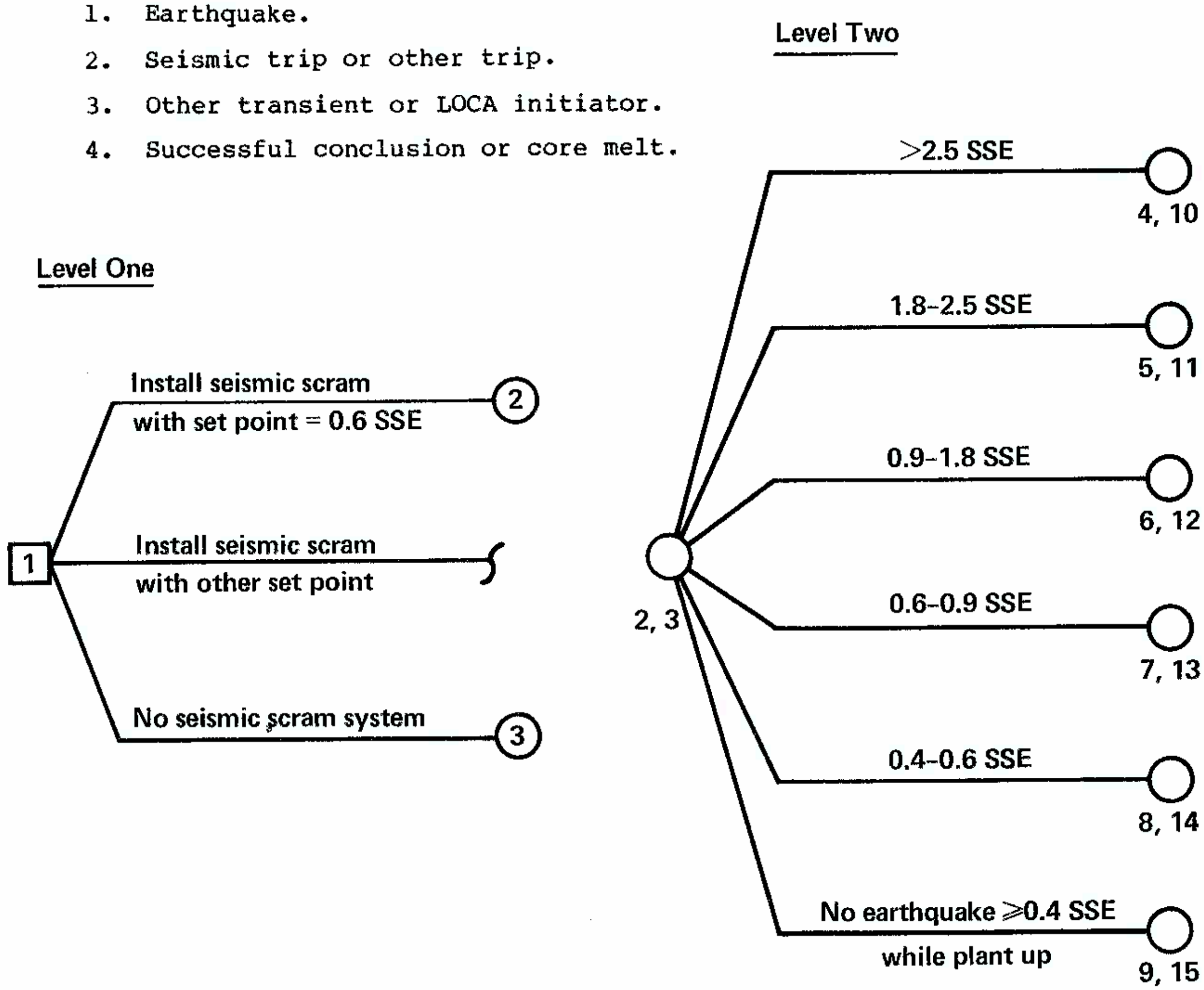
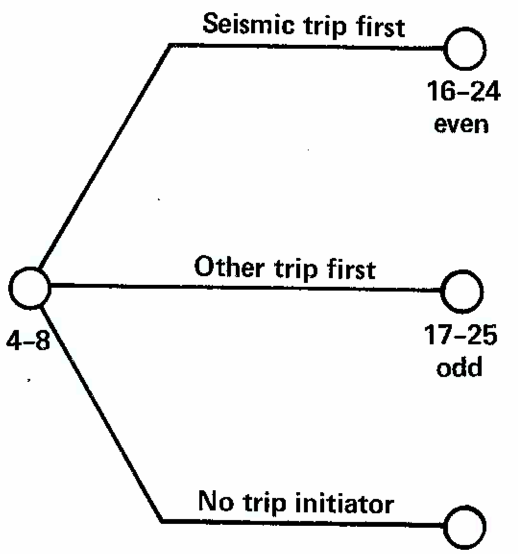
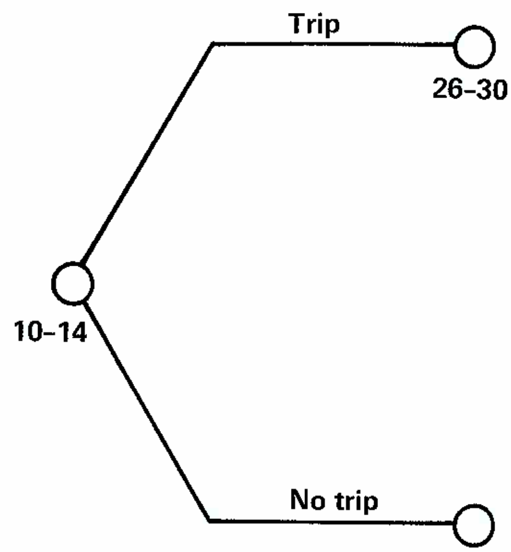


Figure 7.1. Elements in the decision tree, in order of sequential choice and chance forks. Level one: decision for installation of seismic scram system. Level Two: elements corresponding to the occurrence of an earthquake in a given intensity range while plant is operating. Level Three: (next page) elements corresponding to the first initiator of a trip signal. (Left/right corresponds to cases with/without a seismic scram system installed.) Level Four: elements corresponding to the most serious type of transient or disturbance occurring. (Left/right corresponds to cases where seismic trip/other trip has occurred first.) Level Five: elements corresponding to progression to core melt or not.

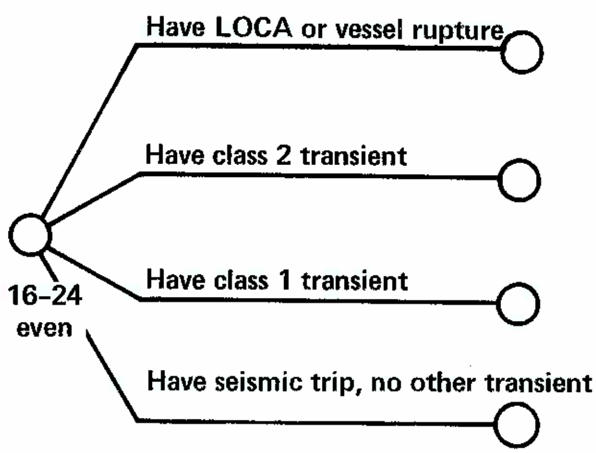
Level Three – with seismic scram installed



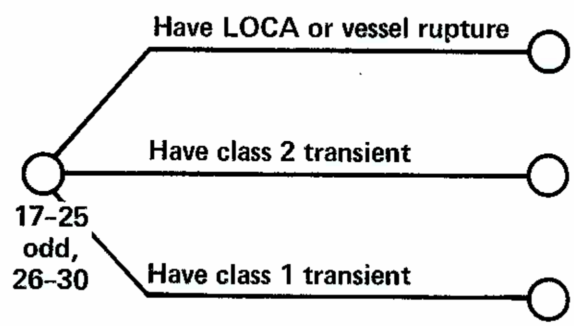
Level Three – with no seismic scram installed



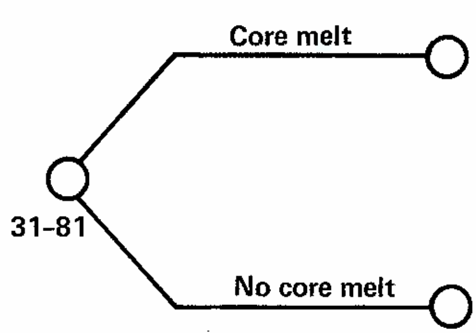
Level Four – if seismic scram trips first



Level Four – if other trip is first



Level Five



We could continue the tree one step further, to containment failure mode, but for this example we end at core melt.

Figure 7.1(a-e) shows the elements of our decision tree: the decision fork, and chance forks for the steps in a possible event sequence. Figure 7.2 shows how these elements are connected. (For clarity, only a few branches are completed.)

The utility of each path in the decision tree will be expressed in the presence or absence of:

1. Core melt.
2. Simultaneous loads on equipment (earthquake and transient).
3. Loss of electric power generation.

7.3 EVALUATION OF THE DECISION TREE

In this section we show how to evaluate the decision tree. Actual numbers would be site-specific. We use input numbers from several sources for illustration. The results would have to be reevaluated for a specific site and plant design. Because some of the input numbers are only poorly known or estimated, we evaluate differences between the two major branches (install seismic scram or not) directly rather than calculating an expected utility for each branch and then taking the difference of the final numbers. Sensitivity of the results to major uncertain parameters is considered.

For the probabilities of events such as offsite power loss or turbine vibration trip versus earthquake intensity in units of the SSE, we are thinking of a Western U.S. region where the site hazard and the design SSE are higher than for most eastern and central U.S. sites. For the probabilities of occurrence of earthquake intensities in units of the SSE (only the relative probabilities are important for the evaluation) we use the Zion values from the SSMRP. For the probabilities of initiating events and conditional failure probabilities, we also use Zion evaluations from the SSMRP.[5]

7.3.1 Probability of Earthquake Occurrence

These are taken from the SSMRP results for the Zion site[5] and are multiplied by 0.7, since we want the probability per year of an earthquake while the plant is operating. The probabilities for the branches are given in Table 7.1.

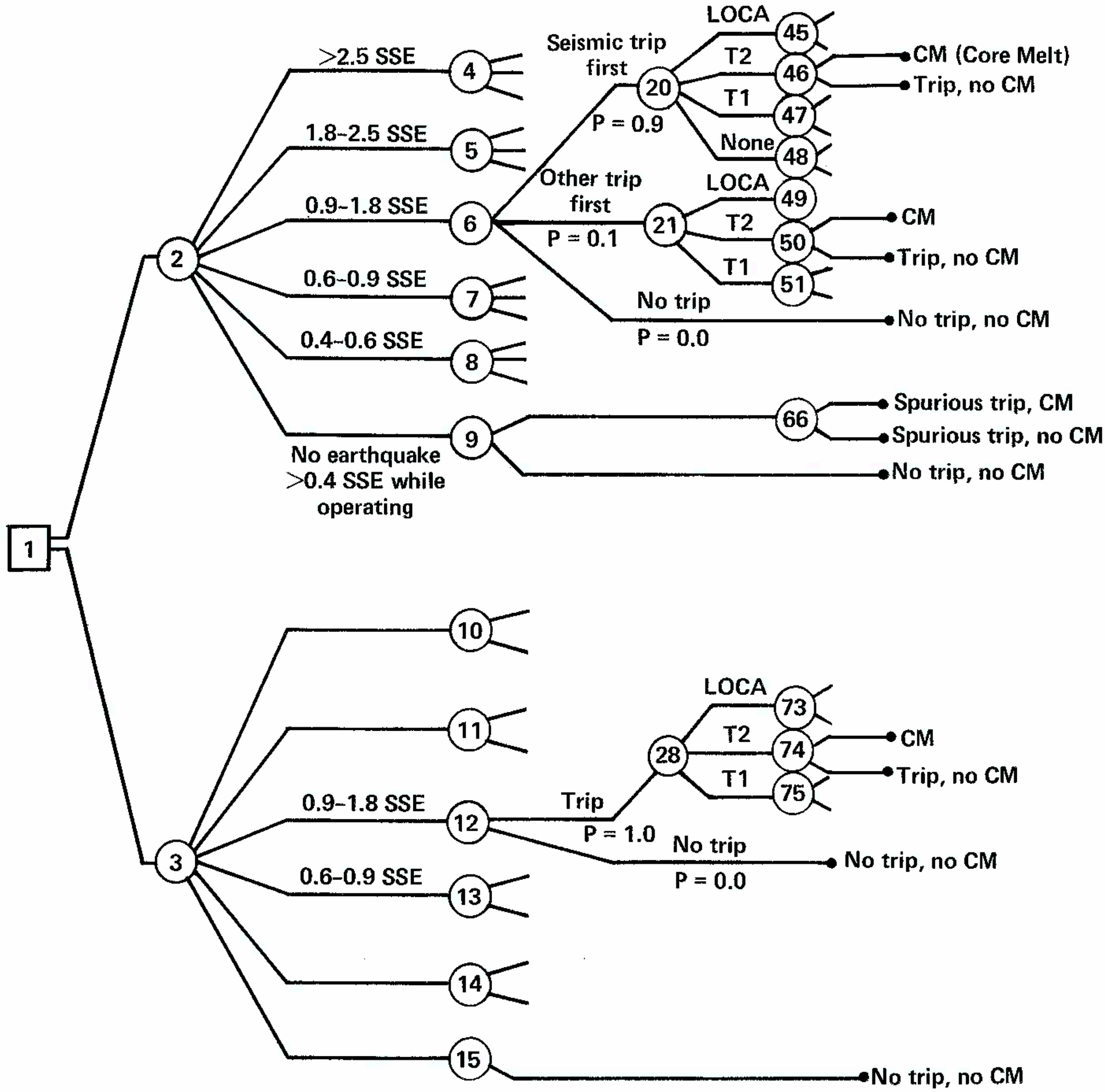


Figure 7.2. The decision tree. For clarity, only a few branches are completed.

Table 7.1. Annual probability of an earthquake at the Zion site while the plant is operating.

Earthquake interval (SSE)	Annual probability $(y^{-1})^{a,b}$
0.4-0.6	8.4E-4
0.6-0.9	4.5E-4
0.9-1.8	2.5E-4
1.8-2.5	1.3E-5
>2.5	2.2E-6

[a] Assuming 70% operation factor.
[b] From Ref. 5.

7.3.2 Seismic Trip First

As discussed in Section 6, a seismic trip will usually give a lead time of 5-20 s before another trip initiator in an earthquake; a lead time of 2-3 s is enough to be significant. We assume that the probability of a seismic trip being significantly early before another trip initiator is 0.9.

7.3.3 Other Trip Initiators

The probability of trip initiators, conditional on an earthquake in one of the intervals, is taken from the SSMRP[5] and listed in Table 7.2. The LOCAs are for the most part small.

7.3.4 Probability of Core Melt

The conditional probabilities shown in Figs. 7.3-7.6 for the cases where there was not a seismic trip before another trip, are taken from the SSMRP.[31] These numbers were from an illustrative calculation rather than a complete one. We will attempt to express conclusions with these results as a parameter.

For the cases with a seismic trip first, we estimate a reduction in core melt probability by about a factor of 4. This also will be a parameter in the results. The reasons for this estimated factor are as follows.

Table 7.2. Conditional probablility of transient or LOCA, given an earthquake.[a]

Earthquake interval (SSE)	LOCA	No transient	T1	T2
0.6-0.9	0.40	0.36	0.24	6E-5
0.9-1.8	0.0	0.40	0.59	2.5E-3
1.8-2.5	0.0	1.9E-2	0.93	5.2E-2
>2.5	0.0	0.0	0.74	0.26

[a] From Ref. 31.

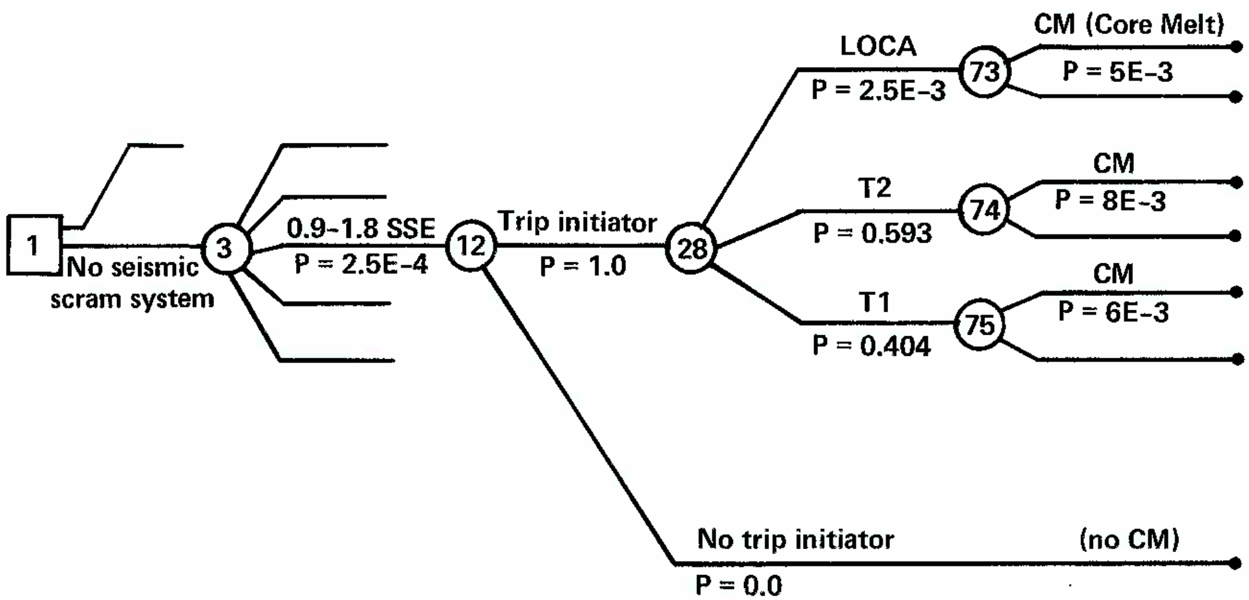
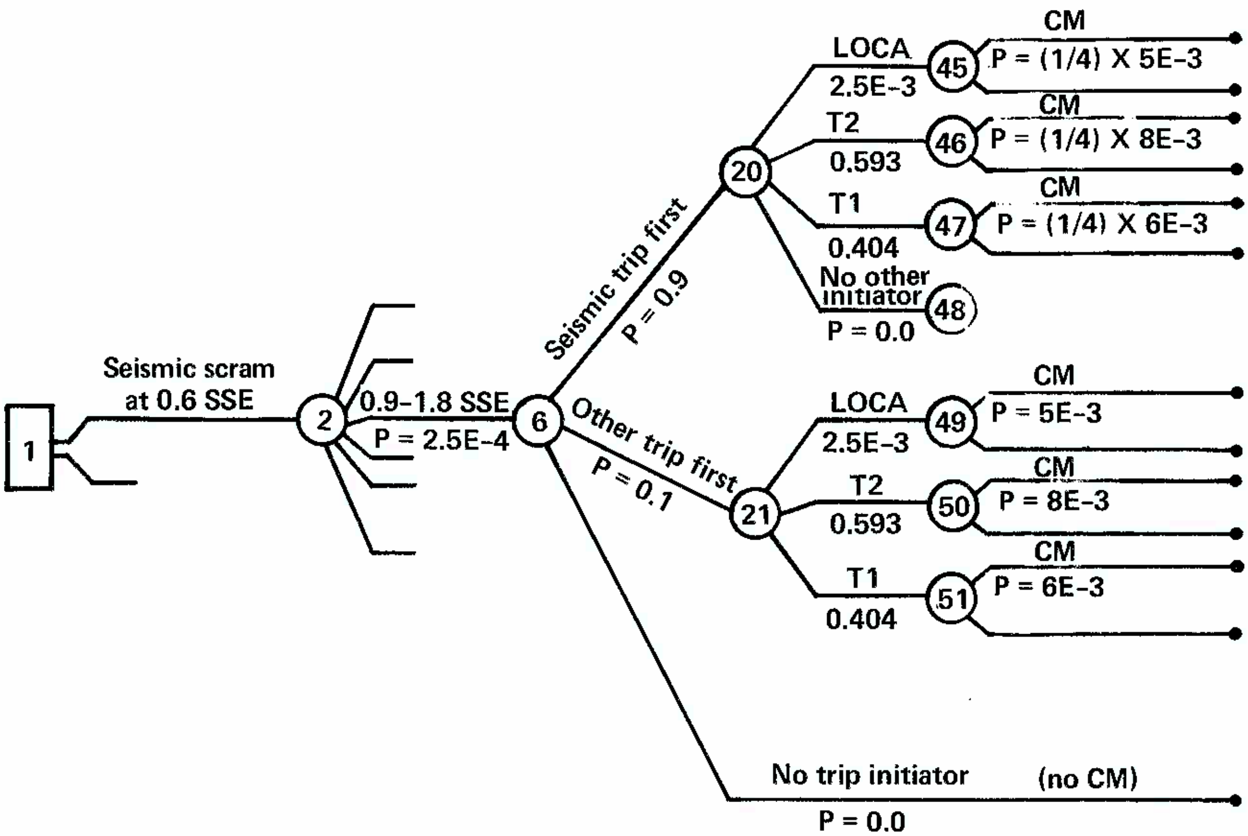


Figure 7.3. Evaluation of the decision tree branches for the case where a reactor with no seismic scram system experiences a strong earthquake (of intensity 0.9-1.8 SSE). The probabilities after the earthquake are conditional probabilities.



For E.Q. range 0.9-1.8 SSE, $\text{Prob (CM)} = (0.9 \times 0.25 + 0.1 \times 1.0) \times P(CM)|_{\text{No seismic scram system}}$
$= 0.32 \times P(CM)|_{\text{No seismic scram system}}$

Figure 7.4. Evaluation of the decision tree branches for the case where a reactor with a seismic scram system experiences a strong earthquake (of intensity 0.9-1.8 SSE).

Among the T2 transients resulting in a core melt, the most likely sequence involves loss of ac power to the auxiliary pumps and loss of the auxiliary steam turbine feedwater system (see Section 6.3). An earlier seismic scram will reduce the pressure transient in the secondary circuit. Hence, we expect that the early seismic scram will significantly reduce the probability of the auxiliary turbine becoming unavailable. The seismic scram will not directly affect the diesel generators or onsite ac power distribution, but it will slow down the accident progress and allow 10-15 min longer for recovery efforts.

Among the LOCA sequences leading to a core melt, the most likely sequences involve a failure of emergency coolant injection or recirculation. In the blowdown phase, an early seismic scram will mean less heat content in the core, hence lower pressure during blowdown (see Section 6.3.2). This

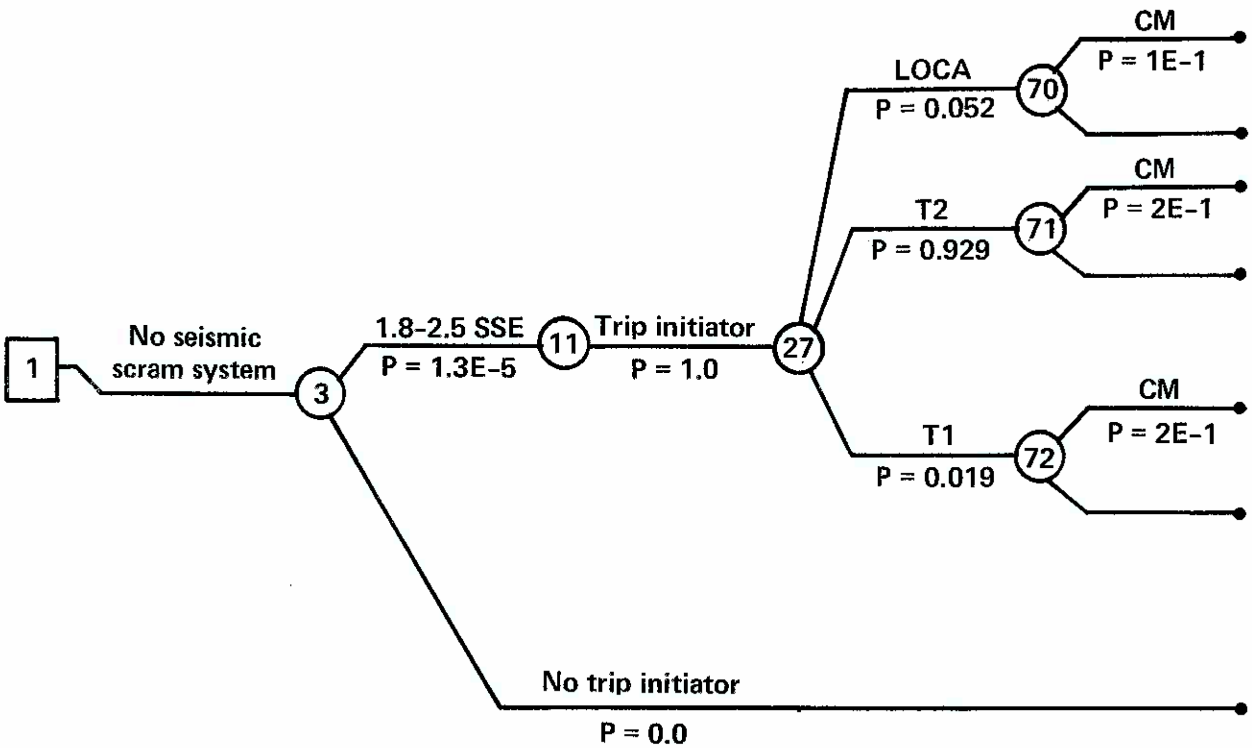


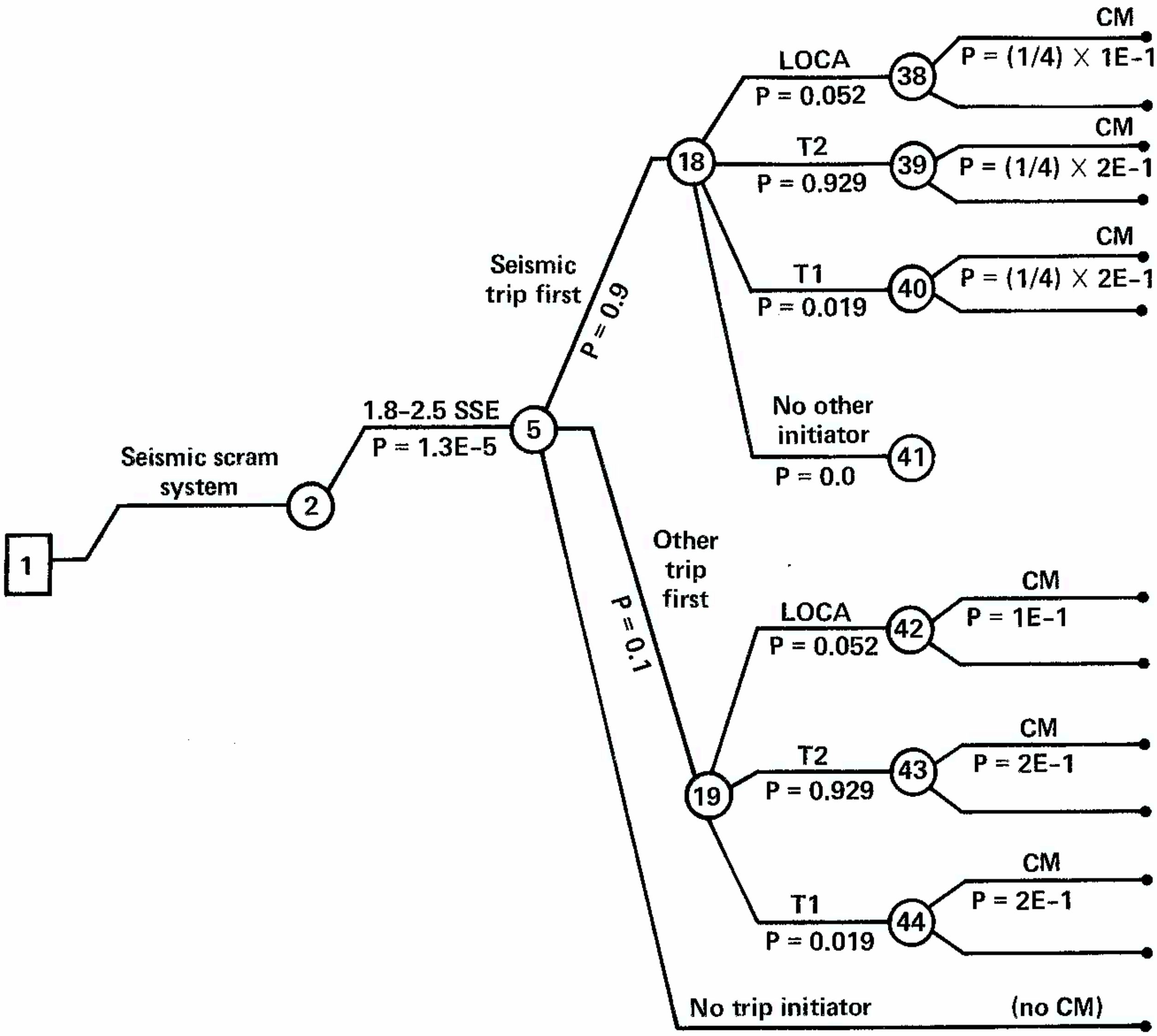
Figure 7.5. Evaluation of the decision tree branches for the case where a reactor with no seismic scram system experiences a very strong earthquake (of intensity 1.8-2.5 SSE). The probabilities after the earthquake are conditional probabilities.

results in (i) earlier depressurization to the level where the injection systems can operate, and (ii) lower rate of blowdown, hence perhaps requiring fewer injection pumps to make up the flow. This effect may be small or large--thermohydraulic calculations are needed to quantify it.

A fraction of the possible LOCAs are very small ones, where the AFW heat removal path is needed. Then the comments above for transients also apply. An overall factor of reduction in core melt probability need not be the same for LOCAs as for transients, but we will assume the same value of 4.

7.3.5 Evaluation

Figures 7.3 and 7.4 show the evaluation for the earthquake interval 0.9-1.8 SSE, without and with a seismic scram system. Each terminal event in the tree has a result of either core melt, or no core melt but a trip, or no trip and continuation of power generation. (The probability of the last alternative is zero here.) The probability of each terminal event can be evaluated by multiplying the conditional probabilities along the path leading



For E.Q. range 1.8-2.5 SSE, Prob (CM) = $(0.9 \times 0.25 + 0.1 \times 1.0) \times P(CM)|_{\text{No seismic scram system}}$
$= 0.32 \times P(CM)|_{\text{No seismic scram system}}$

Figure 7.6. Evaluation of the decision tree branches for the case where a reactor with a seismic scram system experiences a very strong earthquake (of intensity 1.8-2.5 SSE).

to it. Rather than doing that, we note that the 0.9 probability of seismic trip first, together with a factor of 4 reduction in that case, give an overall reduction in core melt probability by about a factor of 3:

$$P(\text{Core Melt})_{SSS} = (0.9 \times 0.25 + 0.1 \times 1.0)\ P(\text{Core Melt})_{\text{No SSS}}$$
$$= 0.32\ P(\text{Core Melt})_{\text{No SSS}} ,$$

where SSS = seismic scram system.

Or, equivalently, we can look directly at the reduction in probability of core melt:

$$\Delta P(\text{Core Melt})_{SSS} = (0.9 \times 0.75)\ P(\text{Core Melt})_{\text{No SSS}}$$
$$= 0.68\ P(\text{Core Melt})_{\text{No SSS}}\ .$$

This equation expresses the risk reduction as directly proportional to three factors: the fraction of cases with significant trip lead time, the fraction of reduction in core melt probability in those cases, and the original risk from seismic events. If the last-mentioned is a significant contributor to the total risk, then risk reduction is significant.

A recent study[32] examined the overall risk at a specific reactor site. It found that seismic-initiated accidents contribute about 12% of the annual probability of core melt, but about 90% of the annual probability of major radioactive release, since the earthquake affects both the reactor and the containment systems. Their earthquake-induced probability of core melt was $6 \times 10^{-6}\ y^{-1}$, with uncertainty of a factor of 6 in each direction.

Figures 7.5 and 7.6 show the evaluation for another earthquake interval. If similar factors of reduction by virtue of the seismic scram system apply as for the other earthquake interval (as we assume), then a similar overall factor results.

Figures 7.7 and 7.8 show part of the evaluation for a lower earthquake interval. Now a disadvantage of the seismic scram is seen--there are cases where there would be no non-seismic trip, and the seismic initiation of a trip is creating an opportunity for core melt. This increased contribution to core melt probability is still much less than the reduction. (A very low trip level would tend toward reversing this net gain.)

7.3.6 Consequences with No Earthquake

A seismic scram system might increase the frequency of scrams by about 0.1 per year. Given such a random scram, the conditional probability of leading to a core melt is 1×10^{-6} or less.

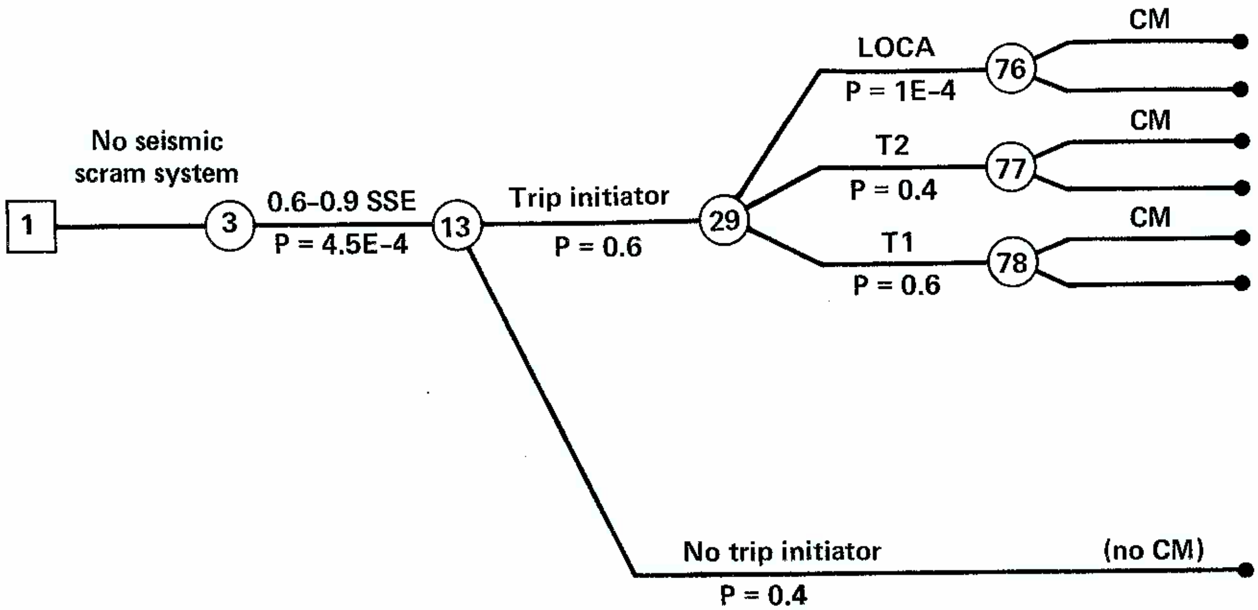


Figure 7.7. Evaluation of the decision tree branches for the case where a reactor with no seismic scram system experiences a medium-intensity earthquake (0.6-0.9 SSE). The conditional probabilities for core melt are unavailable.

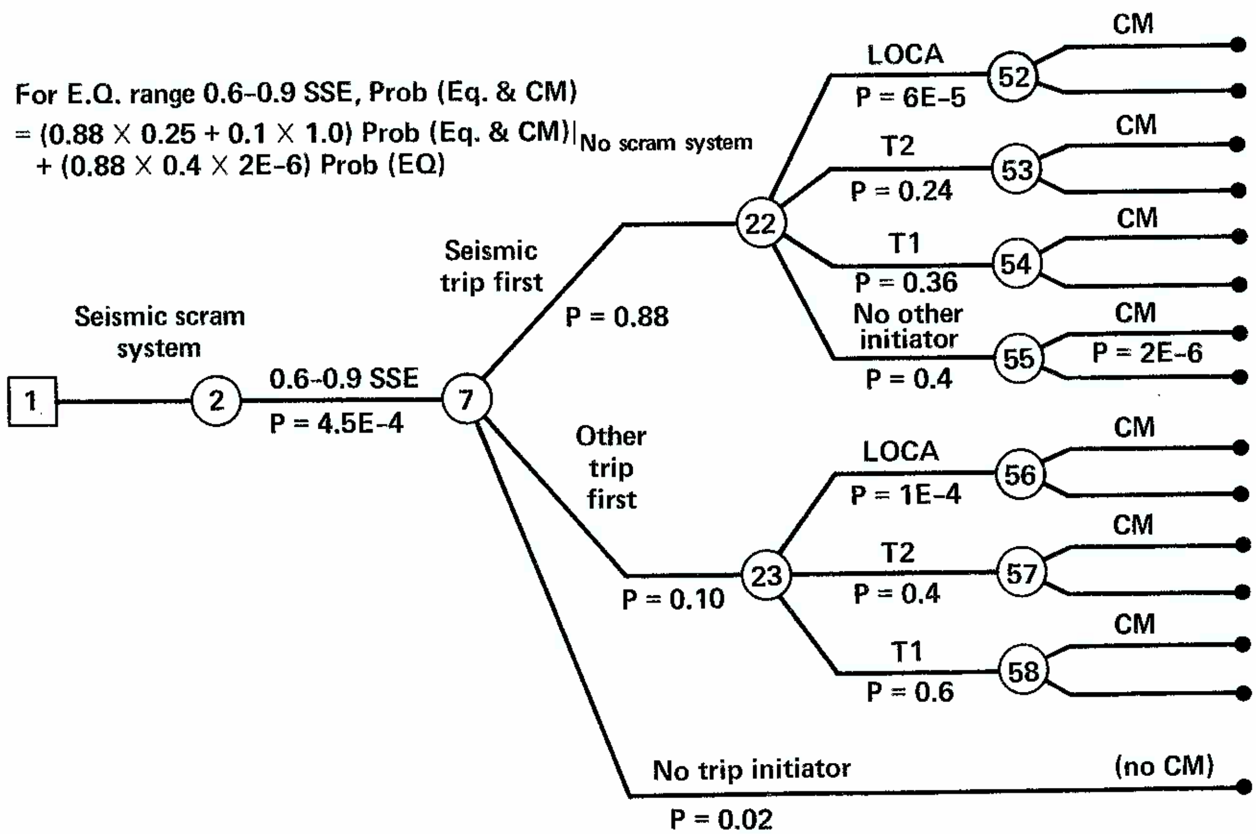


Figure 7.8. Evaluation of the decision tree branches for the case where a reactor with a seismic scram system experiences a medium-intensity earthquake (0.6-0.9 SSE). The conditional probabilities for core melt are unavailable; they are expected to be reduced if there is an early seismic scram.

7.4 SUMMARY

Figure 7.9 summarizes the changes in the probabilities of consequences by installation of a seismic scram system in our hypothetical example plant.

There is an increase in the annual frequency of scram by about 0.1 per year, compared to a scram frequency from other causes of 2.5 per year.

There is an increased probability for loss of power generation in an earthquake emergency, and for simultaneous earthquake and scram loads, of 2×10^{-4} per year. This is somewhat offset by about 5×10^{-4} per year of cases where simultaneous earthquake and transient loads are decreased in magnitude.

There is an increase in probability of core melt without an earthquake of about 1×10^{-7} per year. This increase is more than balanced by a decrease

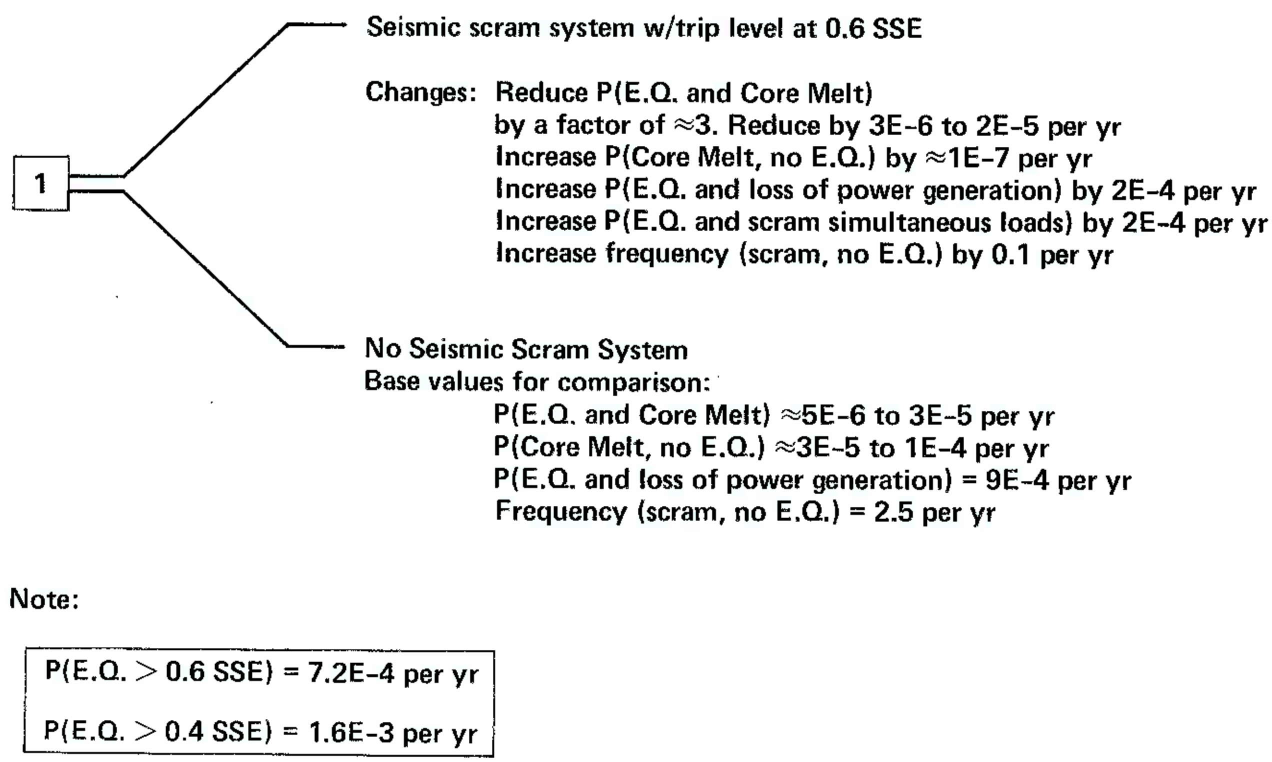


Figure 7.9. Summary of the changes in probabilities of the various consequences affected by installation of a seismic scram system in the hypothetical plant. Values are as qualified in the text.

in probability of core melt with an earthquake by about a factor of 3, or (very roughly) a reduction on the order of magnitude of 4×10^{-6} per year. The reduction is even more significant, since an earthquake affects both the core and the containment protective systems at the same time.

SECTION 8.0 RESULTS AND IMPLICATIONS

8.1. RESULTS

1. A high-level seismic scram system would usually give an early reactor trip signal, usually 5-20 s before another trip initiator.
2. A high-level seismic scram system has advantages and disadvantages. The advantages arise from several factors:
   a. Earlier stopping of heat generation.
   b. Lower heat content of fuel rods and core.
   c. Lower pressure transient.

The above factors lead to advantages:

a. Probability of core melt given an earthquake is reduced.
b. In a LOCA blowdown phase, temporary absence of water cover for the top of the fuel may be reduced or avoided.
c. Simultaneous loads of earthquake and transient are reduced in some cases. (However, in other cases, and for different components, simultaneous loads are increased.)
d. The more likely types of accident affecting the core proceed more slowly (by 10-15 min) toward a substantial damage condition.

The relative probabilities of advantages and disadvantages were summarized in Section 7.4. Some of these items are comparable and some are different. The increase in probability of core melt without an earthquake is very small compared to the decrease in probability of core melt initiated by an earthquake. There are at least qualitatively balancing increases and decreases in occurrence of simultaneous loads. The major net changes (in their separate units) from installing a seismic scram system as estimated for the example plant are

a. A decrease in the probability of core melt after an earthquake, by a factor of roughly 2/3.

b. An increase in the probability of reactor trip following an earthquake, by an amount on the order of 2×10^{-4} per year. This loss of generation affects the power network, hence may affect emergency power for hospitals, rescue services, and other nuclear reactors following the earthquake.

3. The advantage calculated, namely, the net reduction in core melt probability, depends on the following assumptions, which should be further evaluated for any site-specific application:
   a. For earthquakes above the trip level, there is a high probability that some disturbance will eventually trip the reactor without a seismic scram.
   b. The lead time of a seismic trip over other trip initiators is usually greater than the 2-5 s needed to get the physical effects described in Section 7.3.
   c. These physical effects do affect components which have a key role in the event sequences leading to core melt, significantly reducing their probability of failure.
4. Other systems which might trip a reactor in an earthquake, such as turbine vibration, water level, or offsite power, do not provide an assured or rapid trigger to scram in case of a large earthquake. Little analysis or data are available, but effects on these components seem to be highly variable, depending on design factors and earthquake frequency content as well as peak acceleration.
5. One disadvantage, tripping in some earthquakes where a trip would not otherwise be required, may be more serious in the eastern and central U.S. Earthquake attenuation with distance is less there than in the west, and numerous reactors may be tripped, affecting availability of offsite ac power in the emergency to those reactors nearest the epicenter as well as to communities.

8.2 IMPLICATIONS

1. The evaluation of a seismic scram installation is site-specific and design-specific. This evaluation would include some non-nuclear-safety systems, such as the power conversion system and the regional power grid.

2. If a seismic scram system is planned, attention can be given to reducing the disadvantages, which are
   a. The shutdown of a multiunit site tripping the power grid.
   b. Reduced generating capacity overloading the grid (consider selective load shedding).
   c. Dilution of attention to non-seismic safety.
3. The main goal is to reduce the likelihood of event sequences leading to core melt and containment release. Thus, aside from seismic scram, attention should be directed to power sources in the event of an earthquake:
   a. Electrical buses and circuit breakers providing power to key components.
   b. Offsite grid and connections: further hardening, both structural and electrical.
   c. Diesel generators. Consider:
      i) Diversity as well as redundancy.
      ii) Early (low-level) seismic trigger to start one diesel generator (DG).
      iii) Withhold one or two DGs from startup until earthquake strong motion is over, even if needed. (A seismic trip of the reactor would complement this step--less core heat to be removed allows a longer time delay before diesel power for the auxiliary feedwater pump is urgently needed.)
      iv) Stock materials and tools for quick repairs under emergency conditions.
   d. Auxiliary steam turbines: further hardening of controls and power sources.
4. Thermohydraulic calculations for a range of reactor trip times and LOCA sizes and locations would be useful for response planning and for seismic scram evaluation.
5. The sequence of events, and the preferable mitigating and restoring actions, should be planned out for various sequences which could occur in an earthquake, not just for bounding cases.

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