SYSTEMS MANAGEMENT SUPPORT FOR ERCDC STUDY OF UNDERGROUNDING AND BERM CONTAINMENT

INTERIM REPORT

MASTER

**PRELIMINARY PROGRAM ASSESSMENT AND
FOLLOW-ON PROGRAM DEVELOPMENT**

AUGUST 1977

prepared for
California Energy Resources Conservation and Development Commission

prepared by
THE AEROSPACE CORPORATION
Energy and Transportation Division

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Preliminary Program Assessment
and Follow-on Program Development

Prepared for
California Energy Resources Conservation
and Development Commission

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THE AEROSPACE CORPORATION
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EXECUTIVE SUMMARY

1.0 INTRODUCTION

Nuclear safety has been a subject of active debate in the California Legislature since the early 1970s. In the fall of 1975, the California State Assembly Committee on Resources, Land Use, and Energy, conducted a series of 16 hearings on issues related to the safety of nuclear power generation within the state. These hearings called attention to the anticipated prominence of nuclear energy production in California and identified for the legislators the dominant issues in the nuclear safety debate. As a result, the California Legislature produced three bills: Assembly Bills (AB) 2820, 2821, and 2822 (1976). They were written with the apparent objective of slowing down the rate of construction of nuclear power plants within the state until (a) certain problems which were perceived to be "clogging" the nuclear fuel cycle (reprocessing of spent nuclear fuels and waste disposal) were resolved (AB 2820 and AB 2822) and (b) a study was conducted to determine whether it was desirable to locate reactors underground (AB 2821).

Under the mandating legislation (Public Resources Code, Section 25524.3), the California Energy Commission¹ was assigned to perform a detailed study regarding the potential for nuclear power plant "undergrounding" and "berm containment" siting. This legislation became effective on 1 January 1977 and required the Commission to complete "a study of the necessity for, effectiveness of, and economic feasibility of undergrounding and berm containment of nuclear power reactors. . ." At the completion of the study, the Commission must resolve the question of whether to require underground siting of nuclear power reactors.

¹ California Energy Resources Conservation and Development Commission (ERCDC)

Proponents of underground nuclear power plants have identified several features which they feel are particularly attractive. For example, underground siting has been perceived as having the potential for providing an additional safety margin when compared to surface plants. Underground facilities have also been recommended on the basis of the potential for improved flexibility in facility siting, reduced seismic environments and increased resistance to sabotage. Some have also suggested that aesthetic and psychological benefits might be achieved by underground siting.

The incremental cost of underground facilities has been recognized as another important issue. Any decision relative to undergrounding must be heavily influenced by the results of a cost-benefit analysis. Other issues, such as the environmental, political, and sociological impacts of undergrounding relative to surface siting, must also be considered in any in-depth study of the siting question, although the relative priority of these issues could permit them to be deferred until a firm technical basis for an evaluation had been established.

This report presents interim results of the first-half of a study being conducted by The Aerospace Corporation with respect to the technological aspects of the costs and benefits of underground nuclear power plant construction in direct support of the California Energy Commission's legislative mandate in this area. Results of the initial half-year study (January to June 1977) are presented in this report. The program was directed towards problem scoping, methodology evaluation, program definition and planning for subsequent, more detailed investigations of underground facility designs and their potential advantages and disadvantages.

The material presented describes the results of (a) systems analyses which were conducted to determine logical requirements for determination of those elements of a nuclear power plant which should be constructed underground; (b) bounding estimates of incremental plant costs for a variety of underground concepts; (c) applicable prior experience in underground facility design and construction which could be used to identify potential sources of strength and weaknesses of underground nuclear power plants; (d) estimates of seismic environments for underground construction

in California; (e) preliminary descriptions of underground reactor accident scenarios; (f) bounding estimates of the consequences of such accidents, in terms of comparisons of relative emissions of radioactivity with respect to similar accidents for surface-sited nuclear power plants; and (g) results of analyses of several other important technological aspects of the problem.

A description is also provided of the program development work performed to provide planning and criteria for subsequent investigations to determine: (a) definitive underground nuclear power plant designs and costs, and (b) estimates of accident consequences in underground nuclear power plants. These investigations are currently in progress during the second-half "Detailed Study" portion of the Energy Commission program.

2.0 SIGNIFICANT RESULTS AND CONCLUSIONS

The principal design and construction features of a number of underground plant concepts were analyzed. Functional relationships for the principal systems of nuclear power plants constructed on the surface were reviewed. Subject to the constraint of minimizing design changes within the principal hardware elements of a nuclear power plant, a system-level repackaging analysis was undertaken to study the critical relationships between the functional elements of the nuclear facility. The goal of the analysis was to define relationships which reduced structural volumes as much as possible, consistent with operations and maintenance requirements, in order to minimize excavation requirements for underground facilities. Several concepts were developed and analyzed in order to evaluate the cost and safety impacts of alternative design layouts. Various surface or sub-surface locations were postulated for key portions of the nuclear plant systems in these analyses. Rough estimates were made of the costs of the concepts when constructed in underground mined-rock configurations. Somewhat unexpectedly, the design configurations where the majority of the plant systems were placed underground (including the reactor, turbine-

generator, and related auxiliary functions) were found to be relatively less expensive than concepts where only the nuclear steam supply (reactor) related systems were located underground. In the latter case, the costs of long lengths of high-pressure, high temperature steam and/or reactor coolant lines from the underground structures to the surface facilities appear to exceed the costs of the added excavation for fully underground configurations. Moreover, if only the reactor is placed underground, both surface and subsurface containment facilities are required to guard against leakage of radioactive fission products. Consequently, the largest facility costs were associated with the concepts with the most limited underground construction.

Factors contributing to uncertainties in cost estimates were identified for major classes of equipment and labor. Since cost is one of the important elements of any decision analysis procedure, a good deal of emphasis was placed on means of ensuring the accuracy of cost information to be developed during the subsequent, Detailed Study, portion of the overall ERCDC program. Cost accuracy is a function of the level of design detail, therefore, it was necessary to determine which elements of the system had the greatest impact on cost uncertainties in order to determine which future engineering efforts should be emphasized in the subsequent program. The principal design features contributing to the cost uncertainties were judged to be related to excavation and underground construction. These features (and several other less significant elements) contribute to uncertainties of about 15 percent (on the low side) in engineering cost estimates which are based on conceptual designs. Other factors contributing to cost uncertainties, which are not part of the engineering design process, such as (a) uncertainties in prices of goods, services, and money, (b) impacts of labor productivity, (c) regulatory agency effects, (d) schedule uncertainties, etc., could also have large effects on total cost estimates for underground facilities. Many of these latter factors will be investigated in more detail in the subsequent, Detailed Study, phase of the overall ERCDC program.

A limited base of applicable prior experience was found to exist for evaluation of both the costs and effectiveness of underground nuclear facilities. Four underground nuclear power plants have been constructed in Europe. Only one is of relevant size (283 MW_e). The other three are experimental or prototype facilities. No pertinent cost information for design and construction of these facilities was available during the study. Moreover, none of the plants have been designed with the specific objective of minimizing consequences from severe reactor meltdown accidents. However, a partial core-meltdown accident did occur in the experimental plant at Lucens, Switzerland, resulting in permanent closedown of the plant -- but no atmospheric releases of radioactive fission products are known to have occurred.

Underground hydro-electric plant experience demonstrates that large cavities with critical span dimensions in excess of 100 ft have been constructed. The largest nuclear power plant excavations which have been considered would require about 150 ft spans. Consequently, excavation requirements in rock may, in some cases, exceed the current state-of-the-art in underground construction. However, there seem to be no insuperable obstacles to construction of underground plants from a technical standpoint, although specific construction techniques, costs, and the operational characteristics of underground plants remain to be defined in detail.

Seismic criteria for underground construction were developed during this portion of the ERCDC study. Theoretical considerations and experimental data suggest that seismic accelerations at depths in excess of 100 ft are approximately one-half those at the surface in soil, and two-thirds of the surface accelerations in rock. Thus, it appears that seismic effects on underground plants would be less severe than effects for surface plants. This may be viewed either in terms of a reduction in the earthquake hazard for underground plants, or, in terms of the potential for reduced construction costs associated with seismic designs relative to those of a surface sited facility at the same geographic location.

General siting criteria were developed for underground facilities. Suitable rock characteristics were defined in geological terms and rock types meeting these criteria were identified. Based upon these criteria, a review of geologic maps of California indicated that there are several regions of potential interest for underground sites within the State.

Three key modes of containment failure (steam explosions, excessive pressure within the containment, and meltthrough of the foundation base mat by the molten reactor core) were examined. Relative comparisons were made of releases of radioactivity from above ground and underground installations for these three dominant modes of containment failure. The results indicate that in an underground facility the consequences for the failure modes are altered and probably diminished when compared to release characteristics for surface plants. The possibility of breaching the containment by steam explosion launched projectiles is evidently sharply reduced. Failure by meltthrough is a problem, even in the underground environment. However, in a meltthrough failure of an underground containment, the path lengths for fission product transport and release to the atmosphere are lengthened by distances at least equivalent to the depth of burial of the structure; hence, transport times would be substantially increased. Failure by overpressurization was judged to be the dominant and potentially most serious problem for underground facilities. Analyses showed that for this failure mode an analogy can be drawn between an underground containment structure and an oversized, defective household pressure-cooker. Under accident conditions within a sealed underground facility, the decay heat of the fission products could be confined within the structure until destructive pressures were reached in excess of the load carrying capacity of the surrounding soil or rock and containment rupture occurred. Under these circumstances, some release of radioactivity might occur within a short time from the underground facility. This failure mode is referred to in the vernacular as the "pressure-cooker effect." Engineered pressure relief systems were defined for detailed study in the subsequent portions of the ERCDC program, in order to minimize the potential for underground containment failure by destructive overpressurization.

Preliminary analyses were conducted of the transport of radioactive fission-products released through the ground. Evaluation of the migration histories of several important radionuclides suggests that even for release depths as small as ten meters, a large diminution (e.g., perhaps as much as a factor 1/1000) may occur in the overall radioactivity of the fission-products as they diffuse to the surface of the earth. In addition, the results suggest that a significant time delay (e.g., about a month for a gas such as Kr^{85} and 70 years for a chemically active isotope such as Sr^{90}) may be introduced before significant quantities of radioactivity reach the surface. This time delay offers opportunities for mitigating the effects of an underground accident on the population. Just how much of this potential for consequence mitigation can be realized will be determined after more detailed study of possible decontamination or evacuation procedures and their necessity.

In the course of these studies, it has been observed that avenues for fission product transport other than the tortuous diffusion paths through the soil (such as fissures, cracks, and ground-structure interfaces) may be very important. If openings such as these were to occur, then much of the benefit derived from the filtering properties of the overburden could be lost unless specific efforts (e.g., soil/rock treatment, tests, etc.) were made to eliminate the existence (or development) of such cracks and fissures. Special attention will be given to this issue in the Detailed Study phase of the ERCDC program.

Another consideration which has become apparent is that the accident mitigating features identified for underground siting are very site and facility design dependent. If the overlying soil or rock is to contribute to accident control, then particular engineering effort must go into achieving this objective. Engineered systems must be developed to aid in controlling pressures and temperatures within the structures, and built to take advantage of the medium surrounding the underground facility. Subsequent studies will be devoted to designing and estimating the costs of such systems.

3.0 OBSERVATIONS

It is emphasized that the results presented in this interim report are not the final results of the ERCDC program. Though the results presented for some tasks represent essentially completed products for the overall program, investigations are continuing in the principal technological problem areas of underground facility design, costing, and analyses of potential consequences from accidents in underground facilities. No final conclusions have been (nor should be) derived with respect to the absolute desirability or overall feasibility of underground nuclear power plant siting relative to surface nuclear plants prior to the completion of the overall ERCDC program. It is not the intent of the authors of this report to prejudge the ultimate outcome of the ERCDC program. Instead, only the pertinent results of a preliminary overview of the problems of underground nuclear power plant designs have been presented herein.

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1.0 INTRODUCTION

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scenarios; (f) bounding estimates of the consequences of such accidents, in terms of comparisons of relative emissions of radioactivity with respect to similar accidents for surface-sited nuclear power plants; and (g) results of analyses of several other important technological aspects of the problem.

A description is also provided of the program development work performed to provide planning and criteria for subsequent investigations to determine: (a) definitive underground nuclear power plant designs and costs, and (b) estimates of accident consequences in underground nuclear power plants. These investigations are currently in progress during the second-half "Detailed Study" portion of the Energy Commission program. A more complete description of The Aerospace Corporation program plan and its relationship to the overall ERCDC program is presented in Section 1.3 of this report.

It is emphasized that the results presented in this interim report are not the final results of the overall ERCDC program. Though the results presented for some tasks represent essentially completed tasks, effort is continuing in the principal technological problem areas of underground facility designs, costs, and potential consequences from accidents in underground facilities. No conclusions should be derived prior to the completion of the overall ERCDC program with respect to the absolute desirability or overall feasibility of underground nuclear power plant siting relative to surface nuclear plants. It is not the intent of the authors of this report to prejudge the outcome of the overall ERCDC program; only the pertinent results of a preliminary overview of the problems of underground nuclear power plant designs are presented herein.

1.1 BASIC UNDERGROUND FACILITY CONCEPTS

To facilitate presentation of the issues surrounding underground siting, a brief review of the concepts follows. Two basic facility concepts were defined for evaluation: "berm-contained" and "mined-rock" underground facilities. Berm-contained concepts include a spectrum of soil-sited designs, ranging from simply burying a more-or-less conventional

surface-sited nuclear power facility under a soil backfill (as shown in Figure 1-1) to "cut-and-cover" excavated and backfilled configurations (as shown in Figure 1-2) with more specialized designs for underground construction.

In a mined-rock underground facility, the reactor and other related sources of highly radioactive materials would be located in caverns excavated in rock, as shown in the schematic diagrams of an existing Swedish underground nuclear power plant presented in Figure 1-3. Generally, mined-rock installations would be expected to be located at greater depths than berm-contained complexes in order to ensure "cavern stability" (i. e., minimize the danger of cave-ins) during excavation.

The several underground nuclear power plants which have been constructed (all in Europe) have been mined-rock concepts. Although the largest facility is about one-quarter the size of the current generation of U.S. power reactors (e. g., the 283 MW_e SENA/Chooz facility in Ardennes, France), it is reasonable to conclude that the technical feasibility of such construction has been demonstrated. Similarly, although berm-contained structures will necessarily be massive to withstand the loads associated with their earth coverings, their design will probably not require major advances in the state of the art of current engineering construction practices. The existence of the European underground nuclear power plants does not, of course, assure the practicability of all conceivable underground plant concepts. The existence of these plants also does not imply that such construction would be inexpensive nor that their effectiveness as a passive safety system has been clearly demonstrated. The underground construction of the European plants was generally motivated by military defense objectives and/or limited surface site availability considerations. The decisions to construct underground plants were not generally the results of concern over public safety. Such concern was a minor factor when compared with strategic objectives. Consequently, design features which could have enhanced safety generally

Figure 1-1. Surface Plant Covered by Mound

(Auxier and Chester, 1973)

Figure 1-2. Hillside Siting Berm Containment

(Karpenko and Walter, 1974)

Figure 1-3. Cross-Section of the Agesta, Sweden, Underground Nuclear Power Plant

have not been included in the facilities, nor have they been evaluated for the existing European plants.

1.2 BASIC ISSUES

More detailed discussions of technological problems associated with the two basic underground facility concepts defined in Section 1.1 are presented in later sections of this report. As indicated, technological problems do not appear to be the dominant issues associated with underground nuclear power plant construction.

Three fundamental questions dominate the unresolved issues associated with underground construction of nuclear power plants. These questions can be formulated as: (a) what increase in overall safety does undergrounding provide?; (b) what does it cost?; (c) is it reasonable to expect that sites for such concepts could be found in California? In addition to these fundamental questions, many additional factors have significant influence on the need and desirability of underground siting. These include, in part, questions of public perception of risk, the possibility for increased availability of siting areas, environmental impacts, and the impacts on utilities, populations and centers of government. The majority of these latter issues were considered to be beyond the scope of the interim portion of the ERCDC program reported herein. Consequently, the efforts described in this report were concentrated on bounding estimates related to partial resolution of the three fundamental issues cited above.

1.2.1 Accident Sequences, Potential Frequencies, and Consequences for Surface-Sited Nuclear Power Plants

Before expanding on the fundamental issues stated above, the potential modes for severe accidents in surface-sited nuclear power plants will be reviewed below.

When a light-water reactor is first placed in operation, it is loaded with fresh, naturally radioactive uranium-oxide fuel with an aggregate activity of about 150 curies in a typical PWR loading, or about 300

curies in a BWR loading¹. This fresh fuel is only slightly radioactive and can be handled with relative safety. During reactor fueling processes, the fresh fuel customarily is handled by workers using only thin cotton gloves for protective shielding. However, after extended periods of reactor operation for power production, the nuclear fission process which produces the energy for the power plant will also produce large amounts of additional, highly radioactive fission products within the fuel. For example, if a large reactor (of about 1 million kilowatts of electrical power output) is operated at full power for about one year, the radioactivity contained within the fuel in the core of the reactor will increase to approximately 17 billion curies. Though the fission-product sources of the radioactivity are largely restricted to the fuel rods themselves, the energetic gamma rays emitted by the radioactive fission products have great penetrating power. Consequently, heavy shielding is necessary to protect operating personnel from the extremely hazardous radiation emitted by the fuel after removal from the reactor.

If the reactor is shut down, the buildup of fission products in the fuel ceases, and the level of radioactivity in the irradiated fuel decreases as time passes. A very rapid rate of decrease occurs initially in the fission product inventory due to the presence of short-lived fission products. The radioactively decaying fission products also produce a substantial quantity of energy ("decay heat"), which must be dissipated in order to prevent the fuel from overheating. Heat removal systems have been provided for periods of reactor shutdown, such as the Emergency Core Cooling System, to remove the decay heat from the reactor fuel rods.

¹ Pressurized-Water Reactors (PWR) and Boiling-Water Reactors (BWR) are the two types of commercial nuclear power supply systems presently available to U.S. electric utilities from U.S. manufacturers. For a concise description of the operational and safety features of U.S. reactors, a brief, lucid primer is presented in the American Physical Society's Review of Light-Water Reactor Safety, Rev. Mod. Physics, Vol. 47, Suppl. 1, Summer 1975.

At the time of shutdown, the decay heat represents about 7 per cent of the rated thermal power output of the plant. This amounts to approximately 225 MW of the thermal power for a plant with an electrical capacity of 1100 MW, and a corresponding rated thermal output of about 3300 MW at full operational loads. Within 10 seconds after shutdown, the decay heat would decrease to about 5 percent of the rated thermal output, dropping to 2 percent after 10 minutes, and reaching 1 percent after 2.25 hours. Within 24 hours after shutdown, the decay heat would fall to about 0.5 percent of the rated thermal output of the plant. The rate of decay heat production decreases very slowly thereafter, due to long-lived fission products, reaching 0.1 percent of rated thermal power (or about 4 MW) about 30 days after shutdown.

At the end of a one-day period, the total radioactivity of the shutdown reactor has decreased from its initial shutdown inventory of 17 billion curies to about 4 billion curies. Within 30 days, the radioactivity will have decayed to about 1 billion curies. It will diminish to about two-thirds of a billion curies in two months slowly decaying to about one-tenth of a billion curies over a period of a year after shutdown. In a similar manner, the same one-year shutdown period would reduce the rate of production of decay heat from its original 225 MW of thermal power to a thermal output of less than 1 MW. Table 1-1 presents a summary of the radioactivity and thermal power output of an 1100 MW_e reactor as a function of time after shutdown.

The enormous quantities of hazardous radioactive fission products and the large thermal power output of the related decay heat are the primary properties which make reactor accidents of unique and special concern. If a reactor accident should occur, the fuel rods must be kept immersed in water to prevent overheating. If the water bath should be lost by some mechanism, the fuel rods would melt in a very brief period (in about an hour or less), releasing the gaseous and volatile portions of the fission products to the confines of the protective reactor containment structure. An existing (or subsequent) breach of the

Table 1-1. Calculated Radioactivity of 1100 MW_e PWR^a at Shutdown and as a Function of Decay Time^b.

Decay Time, Days	Radioactivity (Megacuries)					Total	Total Thermal Power kW
	Iodine and Bromine Isotopes	Noble Gases	All Fission Products	Actinides	Activation Products		
0	1435	1240	13800	3450	10.6	17250	225000
1	265	221	2890	1330	9.19	4230	17400
5	101	105	1870	432	8.42	2310	9720
15	28.7	29.0	1280	39.7	7.50	1330	5600
30	6.74	4.77	947	9.35	6.40	963	4060
60	0.494	0.784	656	6.32	4.76	666	2350
120	0.00282	0.659	401	5.90	2.76	410	1740
210	0.00000309	0.648	244	5.56	1.36	250	1100
365	0.00000218	0.630	146	5.17	0.614	152	659
1097	0.00000218	0.553	47.3	4.45	0.324	52.0	204
3653	0.00000218	0.353	17.9	3.27	0.132	21.3	67

^aReactor is assumed to be shut down just before refueling after a sustained (293 day) period at a specific power of 37.5 MW/metric ton. The time average specific power over the previous 1100 days is 30 MW/metric ton. The reactor is fueled with 3.3% enriched uranium totaling 82 metric tons of enriched uranium fuel.

^bTable from Report WASH-1250

reactor's containment structure could then release the radioactive gases to the environment. The molten radioactive core of the reactor might also penetrate the foundation of the containment structure, furnishing another pathway for release of radioactivity, possibly to populated areas. More detailed descriptions of the events associated with a core meltdown accident are presented in subsequent sections of this report (e.g., Sections 3.1 and 5.2).

The likelihood of an accident of this severity occurring is widely conceded to be very small, but its consequences could be very serious and substantial. The most detailed estimates of the potential risks associated with core meltdown accidents (Reference 1-1) indicate that the largest accidents (where over 1000 people might be killed) have been predicted to occur with a frequency no greater than about once in a billion reactor years. These estimates of very low accident rates have been the subject of vigorous debate. However, the largest estimates of serious accident frequencies have been no higher than about 1 in 100,000 reactor years.

Estimates of accident rates for less severe, but more frequent accidents, involving core-meltdown related deaths, imply that accidents producing one death per year (from radioactivity-induced cancers) might be expected to occur with a frequency no greater than about 1 in 10,000 to 100,000 years per single reactor (Reference 1-1). Therefore, although accident rates are disputed, it is generally conceded that the frequency of severe accidents would be very low when compared to other societal accident rates (e.g., automobile accidents, fires, dam failures, etc.).

A greater subject of controversy is the acceptance of risks (where risk is defined as the product of frequency and consequences for an accident) associated with nuclear accidents. A wide diversity of opinion exists on the subject of risk acceptability, a topic which is being studied in the second-half of the ERCDC program.

Nuclear power plants are designed to prevent or reduce the potential consequences of major accidents by installation of safety features which include massive protective containment structures completely surrounding the reactor. However, if a core meltdown accident (frequently referred to as a "Class-9" accident) should occur, there are several ways by which the reactor containment capability could fail. As indicated in the report WASH-1400, (Ref. 1-1) the most probable cause of containment failure would be overpressurization of the structure as a result of gases (steam, hydrogen, carbon dioxide, etc.) produced by evaporation of the water from reactor systems and by the interaction of the melted fuel with materials within the containment. Several other failure mechanisms for the containment structure have been postulated, e. g., (a) missile impacts resulting from steam explosions produced by sudden contact of the melted fuel in the reactor core with water, (b) valve or piping failures resulting in loss of containment isolation, or (c) meltthrough of the foundation base mat of the containment structure by the melted core. None of these failure mechanisms was estimated by the authors of WASH-1400 to be of nearly the same likelihood as failure due to containment overpressurization.

1.2.2 Perceived Advantages of Underground Facilities

Underground facilities have been perceived as having several beneficial effects which would tend to reduce (or perhaps eliminate) the consequences to the public from a meltdown accident in a reactor. In a surface-sited facility, there are a limited number of steps which can be taken to reduce the likelihood of containment failure in the unlikely event of a core meltdown accident. The most promising approaches include (a) increasing the design pressure-resistant capacity of the structure, (b) increasing its volume, (c) condensing the steam generated during the accident, (d) filtering and venting the non-condensable gases, or (e) some combination of these approaches

An underground facility can provide inherent benefits with respect to increasing containment structural design pressures. However, perhaps its most significant benefit is that it provides an implicit

means for filtering and containing the most biologically damaging radioactive fission products if containment design pressures are exceeded. In such an event it has been proposed that underground concepts could be designed to allow the gases to escape into the surrounding soil and/or rock, utilizing the inherent filtering capacity of earth materials to achieve substantial reductions in atmospheric releases of fission products. The capability of the ground to provide such beneficial effects has been intuitively demonstrated by the effectiveness of the earth in containing fission products from underground nuclear weapons tests in Nevada and elsewhere throughout the world.

Of course, if the ground were to be used as a filter and trap for accidentally released fission products, the undesirable side-effects of underground environmental contamination must also be examined. However, underground contamination could be a less hazardous problem than direct release of radioactive fission products to the atmosphere if fission product migration in the underground environment could be demonstrably limited. Consequently, the undergrounding of nuclear power plants has been perceived to have several beneficial features and perhaps to represent an ultimate passive safety system, which its supporters expect might eliminate the most significant consequences of a reactor meltdown accident.

If a Class 9 accident should produce a containment failure in a surface-sited nuclear power plant, the local population probably would be ordered to evacuate. Under some accident conditions, only a few hours time might be available to complete the evacuation safely. For nuclear power plants constructed underground, on the other hand, it is shown subsequently in this report that release of radioactive fission products filtered through the soil and/or rocks would be delayed in time, reduced in rate and magnitude of radioactivity, and extended in duration. Thus, the urgency for evacuating a population located within the danger area of an underground installation would be reduced. In addition, with increased time limits for action, other

active measure for mitigating consequences might be employed to further reduce and delay radioactive releases.

1.2.3 Safety and Cost Issues of Underground Nuclear Facilities

In addition to the earth filtering effectiveness described in Section 1.2.2, undergrounding could have several other attractive attributes. As discussed in greater detail in Section 3.3, seismic design requirements appear to be less stringent for an underground facility than they would be for a surface plant. Earthquake-induced motions are apparently reduced for underground structures, even at relatively shallow depths of burial. Moreover, the options for design seem to be enhanced in the underground environment which could allow for minimizing the motions to which equipment would be subjected during an earthquake.

The potential vulnerability of the underground containment structure to external hazards also would obviously be reduced for events such as tornadoes, aircraft impacts, etc. Though the potential risks for surface-sited facilities associated with these events can be limited by careful siting and design; these types of problems could be reduced to negligible importance by underground construction.

However, undergrounding should not be considered as a total panacea to reactor safety or other problems. Phenomena as cave-ins, rock falls, flooding, etc., represent potential new (or possibly enhanced) mechanisms for inducing underground accident sequences. Moreover, it has been asserted that underground construction could actually reduce safety due to potentially limited equipment accessibility which might, for example, increase the fire hazard potential. This might also increase the possibility of component failures, due to problems with operations and maintenance within the facility. In order to assure containment integrity, closure and isolation systems on all penetrating passageways to the surface must function effectively. Closure design for these entrances and egresses would require careful attention. More detailed discussions of design problems are presented in subsequent sections, especially Section 4.

Construction and maintenance costs for an underground facility are obviously vital factors. As indicated, factors for both increased and decreased costs are evident for an underground nuclear power plant, depending on design concept. Excavation costs are obviously unique cost additions for an underground facility. However, it is possible that underground construction costs could be lowered as a result of reduced seismic motions and increased allowable internal design pressures for the underground structures. However, project construction periods are expected to be increased by about a year, or perhaps more, by the excavation processes and the resulting underground spatial constraints on construction practices. This would offset design savings resulting from less stringent need for maximum seismic protection. Mitigation of unique underground hazards to the facility and associated new design criteria also may be expected to result in cost increases.

It is generally anticipated that costs of underground construction will be higher than those of conventional surface facilities; but estimates of additive factors to apply to equivalent surface-siting costs have varied over a large range (from incremental factors of less than 10 percent to factors of almost 2). More discussion of sources for cost increments is presented in Section 4.3 of this report.

1.3 PROGRAM OBJECTIVES/APPROACH

As stated earlier, this section presents in greater detail the relationship and function of The Aerospace Corporation to the overall ERCDC program.

A flow diagram is shown in Figure 1-4 for the complete ERCDC program of investigation of underground nuclear power plants. This report presents the results of The Aerospace Corporation's work on the first seven tasks of the Initial Program Definition portion of the study. The general goals for the study are also outlined briefly on Figure 1-4. In brief overview, the objectives of the effort in the Initial Program

Figure 1-4. Study of Underground Siting of Nuclear Power Plants

Definition period were two-fold:

- a. To clarify the technical issues associated with the cost and effectiveness of the underground concepts, establish bounds for their quantitative evaluations, and determine the availability and effectiveness of methods for analyzing the problems associated with facility design and analysis of accident consequences, and
- b. To develop criteria and statements of work for the major external contracts for conducting the Detailed Study Phase of the ERCDC program.

In the Detailed Study Phase industry contracts are intended to do the following: (a) to conduct a study to identify geological regions suitable for underground siting, (b) to develop, through the aegis of competent architect/engineers, conceptual designs and cost estimates for underground facilities, and (c) to perform an evaluation of the potential consequences to the public of core-meltdown accidents which might occur in an underground facility.

The principal technical objectives of the Aerospace study were to:

- a. Clarify the issues associated with underground facility design, and provide a first-order evaluation for a representative set of candidate concepts for underground facilities
- b. Bound the potential radiological consequences for underground meltdown accidents
- c. Identify and characterize geologic regions within the state which might be technically adequate for underground siting
- d. Develop quantitative estimates of the subsurface seismic environment and its associated impact on structure design for underground facilities

- e. Examine the cost factors related to design of underground structures to aid in directing the architect/engineer efforts of the subsequent phase of the program.
- f. Develop a preliminary evaluation of alternative concepts for improving the safety of surface-sited facilities as a basis for evaluating the relative effectiveness of underground facilities.

The sections of this report which follow present the more important results of the study efforts of The Aerospace Corporation during the first-half of the ERCDC program. In particular, Section 2 describes selected experience in underground construction pertinent to the problems of siting power plants. This includes four examples of underground nuclear plants built in Europe, as well as problems faced in the construction of underground hydroelectric installations. A brief review of other studies of underground nuclear power plant concepts is also given.

Section 3 addresses three major areas of important design requirements for underground facilities. The first is a review of the most severe accident sequences, an identification of key failure modes for the containment, and an analysis of the transport of radionuclides to the biosphere in the event of a breach of the reactor containment. The second is a discussion of the desirable attributes for technically acceptable sites for underground facilities together with a limited general survey which was performed to determine potential regions of interest for such sites, if found in California. The third major subsection presents the results of an analysis of earthquake-induced ground motions and soil-structure interactions for underground facilities.

Section 4 presents the results of a systems analysis of the relationships of nuclear plant subsystems to each other and their impacts on the choice of acceptable underground plant configurations. Two concepts of undergrounding are considered: berm-containment and mined-rock

concepts. A set of criteria has been developed and applied in evaluation of preferred configurations.

In order to determine whether equivalent safety could be achieved by alternative concepts other than underground siting, literature was researched for concepts which had been proposed for augmenting the safety of the current generation of surface-sited light-water reactors. Section 5 presents a description and analysis of four approaches which synthesize the significant features of the concepts surveyed into design concepts for subsequent evaluation. Criteria for evaluating these concepts have been presented. Relative evaluations of the concepts have also been provided. The evaluations are, however, admittedly subjective because of technical uncertainties associated with each concept.

Section 6 describes the effort which was conducted to develop the statements of work and their associated criteria and specifications which delineate the tasks to be performed by associate contractors during the Detailed Study portion of the ERCDC program. Baseline concepts and sensitivity study elements are discussed, together with a presentation of the resulting specifications for conducting the follow-on programs.

Section 7 presents the important conclusions and recommendations resulting from the study.

REFERENCES

- 1-1. USNRC, "Reactor Safety Study", WASH-1400, October 1975.
- 1-2. Union of Concerned Scientists, Joint Review Committee, "Preliminary Review of the AEC Reactor Safety Study", November 1974.

2.0 EXPERIENCE IN DESIGN AND CONSTRUCTION OF UNDERGROUND FACILITIES

As mentioned in Section 1, only four underground nuclear power plants exist. Each was constructed in rock, and therefore pertinent experience and relevant research data are necessarily limited. For mined-rock concepts, certain relevant results can be inferred from underground hydroelectric facilities. However, these plants generally have been constructed underground for reasons far removed from the motives which inspired this study: i. e., maximum possible public and environmental safety. Nevertheless, the pertinent design information that can be extracted from the hydroelectric facilities has been summarized in this section. A few conceptual studies of underground nuclear power plant designs have been undertaken. Again, limitations of investigative data and experience generally precluded effective and conclusive evaluation of such designs for protection against serious core-meltdown accidents. A brief summary of the available literature on the subject has been provided in the following section.

2.1 UNDERGROUND PLANT CONCEPTUAL DESIGN STUDIES

Many papers have been published describing underground nuclear power plant concepts. The theme of most has been the reduction of consequences to public health and safety in the event of a serious accident, although essentially all have stopped short of evaluation of major core-meltdown sequences. Table 2-1 lists the several concepts proposed, pursuant to the natural characteristics of the site selected for a given installation. Also, reference material from which the data were derived is listed in the table. More designs have been proposed for siting in rock than for soil-covered or berm sites.

Where rock containment designs have been considered, they have been similar in concept to an underground hydroelectric power plant. In constructing a reactor facility in rock, the design goals are to excavate a

Table 2-1. Underground Configuration Concepts

Concept	Section 2 References
<u>Berm Containment (Soil)</u>	2-1, 4
<u>Hillside</u>	
<u>Plain</u>	
Surface Mound	2-8
Cut and Fill	2-2
<u>Mined Cavity (Rock)</u>	
<u>Hillside</u>	2-9, 10, 11
<u>Plain</u>	
Dewatered	2-3, 7
Wet Sites	
Single Barrier	2-14
Multiple Barrier	2-5, 6, 13, 14, 15, 16
Water Curtain	2-12

cavity in rock, as for a hydroelectric plant. The reactor and associated equipment would be installed so as to ensure ease of operation and maintenance, and also to provide mechanisms for sealing penetrations and access passages to assure containment of potential releases of radioactive fission products. To achieve these goals, a wide range of problems must be recognized and solved. These include questions of span size, groundwater, pressure management, fission product transport, etc. The Aerospace literature survey has indicated that none of the early investigations has considered all pertinent questions in sufficient depth to ensure complete containment of a core-meltdown accident.

The siting of a nuclear power plant in a rock cavity requires rock formations of sufficient size and quality to contain the facility and assure safe operation of the plant over its lifetime. Finding such formations in sufficient numbers and in acceptable diverse locations may be difficult. By comparison, berm containment apparently poses less constraining siting requirements, since soils with applicable qualities are generally available. Previously conducted studies have not provided any significant insight into the suitability of the geography and geology of California for underground siting.

The berm concept originated with the idea of soil cover over a conventional nuclear power plant. In this concept, the soil serves only as a pressure-constraining material, providing some control and filtering of a release of gaseous and volatile fission products in the event of a severe core-meltdown accident. Fission products leaking from a buried containment structure would be retained or delayed in passage through the soil before release to the atmosphere. The soil cover is a unique feature of the berm concept, because it allows the possibility of "tailoring" the permeability of the backfill to achieve both pressure management and fission product entrapment.

Since the berm concept is normally associated with relatively shallow depths of burial, placement of the turbine generator and other auxiliaries at the surface has been considered to have little direct impact on design and construction. It has been suggested that correspondingly lower costs than rock cavity construction consequently may result from less restrictive design constraints. However, the berm concept requires covered plant structures of sufficient strength to withstand the stress of increased static and earthquake-induced loads from the overburden, with consequent potential for higher construction costs than rock-cavity concepts. Designs proposed to reduce construction costs associated with the berm cover include hillside siting and "cut-and-cover" sites.

2.2 UNDERGROUND NUCLEAR POWER PLANTS

Four underground nuclear power plants have been built in Europe, one each in Norway, France, Sweden, and Switzerland. Features relevant to each of these plants are summarized in Table 2-2. Horizontal access to the underground chamber (excavated in a hillside) was selected as a basic design feature for each plant. Each was constructed in rock, but only the Lucens (Switzerland) plant is sited in sedimentary material; the other three are in igneous rock. No directly relevant experience exists for underground plants designed for berm containment.

In each case a safety benefit is claimed from underground siting. The designers of the Lucens, Agesta (Sweden), and Halden (Norway) plants conducted studies to evaluate the properties of the rock medium with respect to containment of fission products, although protection against complete core-meltdown was not included as a design requirement for any of the facilities.

There are two examples in the United States of nuclear reactors with most of the facilities underground. The first of these is Shippingport (Ref. 2-17), a 100 MW_e PWR which became operational in 1957, and the second is Humboldt Bay (Ref. 2-18), a 70 MW_e BWR which became operational in California in 1963. Both reactors are buried at shallow depth so

Table 2-2. Underground Nuclear Power Plants

Plant Name (Location)	Type ^a /Size/Purpose	Motivation for Undergrounding	Maximum Cavity Span, ft/Siting Geology
Halden (Norway)	SGHWR/20 MW _e /experimental	Containment economics	30/gneiss with clay
Chooz (France)	PWR/283 MW _e /power	Military threat	18.5/rock
Agesta (Sweden)	SGHWR/20 MW _e /heating and power	Safety	16.5/granitic-gneiss
Lucens (Switzerland)	HWGC/8.3 MW _e /experimental	Safety	18/sedimentary

^aPWR = Pressurized Water Reactor
 SGHWR= Steam Generating Heavy Water Reactor
 HWGC = Heavy Water Moderated Gas Cooled

that only the tops of the containment structures are exposed at ground level. Although the original objective for undergrounding these plants was use of the surrounding soil as a radiation shield during normal operation, the soil might provide some limited secondary benefits in containing or restricting the release of fission products in the event of a serious accident.

2.2.1 Halden (Norway)

The earliest experience in underground reactor containment is the Halden Reactor Project. In this project, conducted as part of a research program under the auspices of the European Nuclear Energy Agency, a 20 MW_t boiling heavy water reactor plant was built at Halden, Norway (Ref. 2-19). The plant, designed for experimental development of heavy water reactor technology, consists of the reactor, waste pit, heavy water storage tank, fuel element storage pit and heat rejection system, all contained within a single underground rock chamber. No provision is made for electrical power generation by the facility. The underground chamber was excavated in gneiss rock with a roof thickness of about 50 m. Isolation of the cavity from the entrance tunnel is achieved by means of an air lock and interlocking closures. The design dynamic overpressure for each closure is 3 g/cm² (42.6 psi) to accommodate a sudden rupture of the pressure system. The maximum static design overpressure has been calculated to be about 2 g/cm² (28.4 psi). The closures consist of two 1 m thick concrete plugs, with appropriate penetrating access doors, located 7 m apart. Cooling of the cavity in the event of an accident is achieved by means of ventilation and water sprays. For the Halden reactor, a pressure release system was designed, where the containment atmosphere would be released after an accident through a water lyre¹ in a pond.

2.2.2 Agesta (Sweden)

The Swedish nuclear power station located in the city of Agesta is the first underground nuclear facility to generate electric power (Ref. 2-20). This station became operational in 1963 and was operated until 1974, when it was shut down. The reactor is a pressurized heavy water unit rated

¹ The term "lyre" refers to a form of perforated distribution manifold through which the containment gases would be released (Ref. 19).

at 60 MW_t and 10 MW_e . It was decided to locate the Agesta plant, Sweden's first power reactor, close to this densely populated suburb of Stockholm in order to use a part of its thermal capacity for providing steam heat to apartment residences. Because of concern about the effect of an accident on the population, the decision was made to construct the plant underground to take advantage of the natural protection afforded by such construction. Although subsequent nuclear plants in Sweden have been constructed above ground, an effort to study the design features, benefits, and costs of undergrounding nuclear plants in Sweden has continued.

The Agesta plant is constructed in a rock cavern of crystalline rock, probably gneiss or granite. The reactor and supporting auxiliary equipment are located in a steel-lined rock cavern, while the electrical generating plant is located outside of the cavern in a separate building. Steam and water connections between the turbine-condenser and the reactor are channeled to the reactor chamber through tunnels. Heavy steel doors isolate the reactor chamber from the access tunnels, and fast-acting valves are designed to close ventilation ducts automatically in the event of an accident.

2.2.3. Lucens (Switzerland)

The Lucens plant is located on the western side of the Valley of the River Broge (Ref. 2-21). Construction began in 1962, and the reactor went critical in 1965. The reactor is a heavy-water moderated, pressure tube type of 30 MW thermal output. The greater part of the plant, including reactor, turbogenerator and fuel element storage pond, is located underground. The maintenance building, radioactive effluent treatment and switchgear facilities are located outside. The reactor and auxiliary equipment chamber excavations are in sandstone. Depth of cover over the reactor chamber is about 60 m with about 30 m cover over the machinery building.

According to a description of the plant, the underground location of the reactor and its ancillary equipment had an important bearing on the design of the containment (Ref. 2-22). The mechanical properties of the

rock reduced the structural costs of the buildings with respect to those typically encountered in other power stations.

Although the reactor cavity at the Lucens plant is not steel-lined, as are other plants, the containment requirements were met with linings of porous concrete, alternate layers of bitumen and aluminum foils and reinforced concrete. This design was intended to control the effects of the maximum hypothetical accident, defined as the complete rupture of the thermal circuits, meltdown of all fuel elements, and energy released from exothermic (heat-liberating) chemical reactions. The containment was functionally designed to control the release of radioactive fission products into the porous sandstone rock by means of remotely controlled valves. It was expected that temporary retention of fission products in the rock would permit the radioactivity to decay to a safe level before the eventual escape of the fission products into the atmosphere.

Several years ago, the Lucens plant experienced an accident that involved a partial meltdown of the fuel elements. Specific information on the performance of the plant in containing the accident has not been obtained. Since the accident, the plant has been shut down. Ultimate plans for disposal of the facility are unknown.

2.2.4 SENA (Chooz, France)

The largest of the four known underground nuclear power plants is in the Ardennes near the community of Chooz. The reactor is of the pressurized light-water type. This plant is designed to produce electric energy and is rated at 283 MW_e. The reactor is a Westinghouse PWR design, of European manufacture. General construction of the plant began in 1962, and low power operation was initiated in 1965. The Chooz reactor is located in a chamber excavated in rock, similar to hydroelectric plant construction. The reactor and auxiliary chambers are steel-lined and contain the basic equipment for operation and refueling of the reactor. Three cavities, two large and one considerably smaller, house the reactor and steam generators, waste treatment and fuel storage, and emergency equipment in the smallest of the three. The turbine-generator and switchyard are located outside the underground chambers.

The decision to locate the plant underground was two-fold:

(a) strategic (maximum protection against military threat and/or sabotage) and (b) population safety [minimum potential exposure to fission products in the event of a severe (Class 8) accident]. While the exact type of rock in which the facility was excavated is not known, it is believed that the material is a crystalline type of igneous rock, such as granite. Cover depths are also not precisely known, but are estimated to be about 200 ft.

At 283 MW_e, the Chooz plant is sufficiently large to be considered a significant commercial power-producing facility. The plant has had a power production history which compares favorably with surface-sited units of similar characteristics. For the past several years it has had an outstanding performance record, with plant capacity factors in excess of 80 percent.

2.3 UNDERGROUND HYDROELECTRIC PLANT EXPERIENCE

To supplement the limited underground nuclear power plant experience, a brief survey was made of pertinent underground hydroelectric plant construction experience. Though decisions for underground construction of hydroelectric plants usually have been based upon economic considerations rather than public safety, the construction experience has contributed to advancing the state of the art in designing and building large underground structures.

The concept of underground power generation machinery for hydroelectric plants was first introduced about 60 years ago. By 1959, there were about 300 such stations in service throughout the world, with a total installed capacity of 31,000 MW (Ref. 2-24). As of 1963, Italy had 60 plants in operation, Sweden had 20, with more under construction, Norway had 26, and the remainder are located in other European countries. Only a few underground plants are located in the U.S.A. The decision to construct the electrical generation facilities for a hydroplant underground is generally based upon economics, compared with surface hydrostations in similar locations. Furthermore, scenic environments may be less impacted by

underground construction, and the power plant is protected against damaging effects of landslides, avalanches, heavy snow and rainfall.

The dimensions of the power house depend upon the size and number of generating units, transformers, and the hydraulic pressures involved. Cavern dimensions for several representative plants are shown in Table 2-3. In the earlier designs, transformers were installed on the surface, although the present trend is to place them underground. In some cases, as in Switzerland where conditions are favorable, even the high-tension switchgear is underground. In hydroelectric facilities, plant flooding is a major design consideration. For safety reasons, the main water inlet valves are usually installed in a separate cavern parallel to the generating unit cavern, with a provision to drain any leakage water or flood water due to penstock failure into the tailrace. In general, the thickness of overburden above the power house cavern is recommended to be at least twice the width of the excavation with a minimum recommended depth of about 35 ft.

Figures 2-1 and 2-2 illustrate basic configurations and typical dimensions of underground hydroelectric plants. The diagrams also show variations in the design of chamber linings and arch support. The control room generally is located underground with proper remote operating gear, so that the low-tension electrical generators and high-tension switchgear can be operated from underground. Duplicate means of entrance and exit are desirable. Cable ducts provide convenient emergency exits. At installations with low hydraulic head designs, the entrance is through an inclined tunnel or a vertical shaft, but in most high-head plants a regular entrance, as horizontal as feasible, is usually provided for more attractive economy of maintenance and operation.

The additional hazards of an underground facility must be noted. In 1927, an underground hydroelectric plant near Fairfax Falls, Vermont, failed due to a cave-in. This plant had been in operation since 1904. Although the complete details of the accident are not precisely known by the

Table 2-3. Representative Data on Underground Hydroelectric Plants
With Large Cavern Spans

Name/Location	Principal Dimensions of Generator Caverns ft			Rock Type	Remarks
	Width	Height	Length		
Helms/California	83	144	336	Granite	Pumped storage, under construction
Haas/California	56	100	176	Granite	Operating since 1953
Churchill Falls/Canada	84	145	1000	Granite	1000 ft rock cover
Kitimat/Canada	84	139	700	Granite	
Portage Dam/B. C.	67	144	890	Sandstone, Shale	Poor rock condition, concrete roof arch provided, 200 ft rock cover
Marrow Point/Colorado	47	134	206	Schist, quartzite	400 ft rock cover
Poatina/Tasmania	45	84	300	Mudstone, shale	Rock bolts, mesh and gunite lining, 500 ft rock cover
Darquinah/Algeria	106	85	239	--	
Santa Giustina/Italy	93	--	200	--	
Santa Massenza/Italy	95	92	650	--	
Premadio/Italy	85	98	264	--	
Cavergno/Switzerland	92	--	91	--	
Peccia/Switzerland	92	--	--	--

High Head Horizontal Entrance Plant

Very High Head Horizontal Entrance Plant

Figure 2-1. Underground Hydroelectric Plant Configurations
(Dimensions in Meters)

Medium Head Plant

Low-Head, Vertical Entry Plant

Figure 2-2. Medium and Low-Head Underground Hydroelectric Plant Configurations

author, the failure was evidently caused by surface flooding, which caused the shallow rock roof to fail, destroying the plant. This reported incident is the only known cave-in failure of a large U.S. underground hydroelectric facility. Nevertheless, it highlights the significance of the underground structural cave-in as a failure mechanism clearly unique to underground facilities.

Plants have been built in all types of rock formations in different countries. In North America, the installations have been in igneous, granitic-type rocks. A major problem with these plants was caused by the unstable nature of the rock, which sometimes led to excavation problems and, occasionally, fault zones were encountered intersecting the chamber axis. In excavations elsewhere in the world, such as Yugoslavia (Ref. 2-25), where limestone rock is abundant, the high permeability of the material has often necessitated continued pumping of seepage water. At times, the highly pervious rock presented a potential danger of underground floods during excavation. Consequently, certain plant layout concepts, sometimes referred to as the "Swedish" type, with caverns directly below the reservoir, are not generally favored.

2.4 PERTINENCE OF PRIOR UNDERGROUND FACILITY DESIGN AND CONSTRUCTION EXPERIENCE

Studies of potential designs for underground nuclear power reactors have been ongoing for almost two decades. Investigators in the United States and several European countries have attempted to answer the questions of the technical feasibility, cost and accident mitigation performance for large (1000 MW_e) facilities of this type. The results of these studies generally confirm the technical feasibility of rock cavity and berm construction, although emphasis of the studies has been on designs in rock cavities. Both experience and conceptual design studies appear to support the conclusion that there will be no major technical obstacles to the construction of underground power reactors.

Past studies of underground reactor plant feasibility frequently conducted accident analyses as they do in a surface plant, i. e. limiting their analyses to Class-8 accidents without specific attention to the problems of a serious (Class-9) reactor core melt. The differences and complications that are expected in designing a plant to cope with the far more difficult task of minimizing the consequences of a Class-9 accident, limit the relevance of most of the previous studies and experience in plant design.

Nevertheless, the construction and operating experience of underground nuclear plants is pertinent, particularly the experience at the Chooz plant. At Chooz, the Nuclear Steam Supply System was accommodated without redesign. Moreover, the underground facility has apparently achieved operational and maintenance standards which compare well with similar surface plants.

The experience of underground hydroelectric facilities, suggests potential problems in cavity flooding and cavity construction, which may contribute to cost increases for underground facilities. Adequate attention must be given to such issues in subsequent plant design efforts. These issues need to be recognized and included in investigations of the plant design and site property specifications, as discussed in Sections 3.2 and 6 of this document.

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3.0 DESIGN REQUIREMENTS FOR UNDERGROUND NUCLEAR POWER PLANTS

Before detailing the basic underground design concepts, several critical design parameters which have major influences on concept configurations are presented. The effectiveness of burial in reducing fission product releases to the environment has been analyzed and results are presented for expected reductions in total releases and changes in time of delivery for transport of radioactivity to the atmosphere. Siting characteristics and the general availability of suitable geological materials within the State of California have also undergone preliminary analysis. The final subsection presents the results of an analysis of the effects of underground construction on seismic environment and loadings.

3.1 PLANT ACCIDENTS AND PUBLIC SAFETY

In this section, the results of selected core-meltdown accident sequences are reviewed with respect to their impact on public safety. The accident sequences of particular interest here are only those leading to a major core-meltdown (or Class-9 accident). Calculations have been performed suggesting that an accident of this severity is highly unlikely. (Ref. 3-1) Whatever the precise probability of a core-meltdown accident, one of the major motivations for conducting this study of underground siting was to investigate the effectiveness of the concept in mitigating the effects of a Class-9 accident. Therefore, in this section the question of the probability of occurrence has been deferred in order to address directly the nature of the Class-9 accident and to relate its consequences to underground and surface-sited nuclear plants.

3.1.1 Containment Failure Modes and Core-Meltdown Accidents

The literature on the general subject of underground siting of nuclear power plants is already extensive, although little contains

directly relevant material for evaluation of siting effectiveness with respect to Class-9 accidents. Of the several conceivable events capable of initiating a Class-9 accident, the sequence of events generally considered in the literature is described in Figure 3-1. The sequence begins with a design-basis, loss-of-coolant accident (LOCA) initiated by a large, double-ended, "guillotine" break in the primary loop piping of the nuclear reactor steam supply system. Water and steam are expelled from the reactor pressure vessel and "blowdown" is complete in a matter of seconds. The resultant loss of the water moderator within the reactor's nuclear core stops the chain reaction, but considerable energy remains in the fuel in the form of the decay heat of the radioactive fission products. Next, the generated steam and gas pressurize the containment and the engineered safety features (ESF's), such as the emergency core cooling system (ECCS), are activated. At this point, if the ESF's perform as intended, the sequence is stopped and the accident is contained. This is termed a "Class-8 accident". On the other hand, in the unlikely event that the ESF's malfunction, the sequence can escalate to a Class-9 accident.

3.1.1.1 Loss-of-Coolant Accident (LOCA) Without Effective Emergency Core Cooling

A probable sequence of events describing in greater detail the core history during a large LOCA without effective emergency core cooling can be divided into seven periods (Ref. 3-2). It should be kept in mind however, that the following description is uncertain in many respects for the physical processes involved are not thoroughly understood. Thus, at many points in what follows, plausible estimates of prevailing conditions have been given.

Period 1: Blowdown

The double-ended rupture of a large primary system pipe produces a violent expulsion of most of the water in the reactor pressure

Figure 3-1. Possible LOCA Sequence with Massive LWR Core Melt

vessel into the containment. This event is termed "blowdown". Three features of the blowdown influence subsequent core behavior. These are the (a) blowdown forces, (b) fuel element cooling during blowdown, and (c) rate of escape from and the quantity of water remaining in the reactor pressure vessel after blowdown.

Current estimates presume that (a) the blowdown forces will not seriously disrupt the core, (b) the fuel element cooling during blowdown would result in removal of about one-half of the stored thermal energy in the fuel rods, and (c) about 10 percent of the water originally in the reactor vessel would remain after blowdown. At the completion of blowdown, which takes at least 5 to 10 sec for the largest breaks, core temperatures may range from about 600°F (320°C) to about 1500°F (820°C) at the hottest section of the core. However, these estimates are quite uncertain, as discussed in the following paragraphs.

Period 2: Core Heatup - Decay Heating Only

Core temperatures begin to rise after blowdown, because of decay heat, at rates of approximately 20°F/sec (11°C/sec) for PWR's and 12°F/sec (6.7°C/sec) for BWR's at the hottest portion of the core. If they start from 1500°F (920°C), local core regions could reach temperatures greater than 2000°F (1100°C) within 30 to 50 sec. Then an exothermic Zircaloy-steam reaction between the Zircaloy fuel rod material in the steam-filled pressure vessel would begin to become a significant energy source. It is therefore important that the ECCS operates successfully before this critical juncture, if possible. Delayed application of the ECCS coolant would be considerably less effective in the presence of a substantial Zircaloy-steam reaction; in fact, it could tend to exacerbate the situation.

Period 3: Core Heatup - Decay Heating Plus Chemical Heating

If portions of the core should reach temperatures greater than 2000°F (1100°C), the Zircaloy-steam reaction would become increas-

ingly significant. The rate of this reaction would initially increase parabolically with time (i.e., doubling the reaction time increases the reaction rate by a factor of 4). As cladding temperatures increase, the pressure vessel may tend to dry out, with a consequent decrease in the reaction rate, as the reaction rate is dependent upon the availability of steam.

The time required to reach the melting point of the cladding, 3360°F (1850°C) at the hottest portion of the reactor in the absence of effective cooling, could be approximately 1 min. However, the beginning of localized clad melting may not mark the effective beginning of core collapse. Fuel rods would have to reach temperatures greater than 3360°F (1850°C) before substantial clad dripping or fuel rod slumping could occur.

Period 4: Core Collapse and Collection as a "Heap" in the Reactor Vessel

The time from the pipe rupture to the collection of the core as a molten heap in the bottom of the reactor vessel is not known precisely. The calculation is highly dependent upon the model chosen for core meltdown and the process chosen to describe the weakening or melting of the internal core structural grid plate. Two outcomes have been considered. Gradual melting of the fuel could result in the slow deposition of the core in the lower portion of the pressure vessel, while an abrupt failure of a core support grid could result in sudden dumping of a major portion of the molten core to the bottom of the pressure vessel.

A time range from 10 to 60 min from pipe break to collection of the core melt as a heap in the bottom of the pressure vessel appears to cover the uncertainties in the models. The lower limit derives from a heat accumulation consideration. With extensive metal-water reaction (50 percent of cladding) and the assumption that all accumulated decay heat goes into heating the fuel, there is sufficient energy to bring the entire core to the melting point of UO_2 within 10 min.

The actual process of a core collapse and collection in the bottom of the reactor vessel is likely to be a much more complicated matter than the description given above. It involves (a) combination of metals in the melt, with substantially altered melting characteristics from the formation of eutectics; (b) dripping molten flow, and resolidification; (c) blockages of flow, steam production, chemical reactions, and (d) many other effects.

Period 5: Pressure Vessel Meltthrough

The nature of a meltthrough at the bottom of the pressure vessel is somewhat uncertain. As noted above, it is not known whether the fuel would dribble through the grid plate slowly as the fuel melted, or if it would form puddles and melt through suddenly in a series of delayed action falls.

From 50 to 80 percent of the molten fuel might reasonably be expected to fall into the pressure vessel bottom. However, this percentage of the fuel would include that portion of the core having the greatest energy for decay heating.

The first effect of the core materials falling into the vessel bottom would be to evaporate rapidly any water remaining in the vessel. A rapid production of steam might affect the flow of core material through the grid plate, but steam explosions (resulting from sudden, hyperintensive interactions of water with the molten core) could blow molten core material up and out of the pressure vessel.

However, when the molten core material has collected in the bottom of the pressure vessel, radioactive decay and possibly some further metal-water reactions would result in continued heating of both the fuel and the pressure vessel bottom.

A conservative estimate of the weight of core material plus vessel internals, about 250 tons, would produce forces in the vessel of

only a few percent of those due to normal operating pressures; hence, the vessel bottom would not be apt to fail until its melting point were approached.

The equilibrium temperature of the melt is somewhat ill-defined due to the uncertainty in the composition of constituent materials. However, temperatures of the order of 4700° to 4900°F (2000° to 2200°C) are probable. Under these circumstances, a meltthrough of the pressure vessel could be expected to occur from 20 to 60 min after core collapse (30 to 120 min after pipe rupture), if there were no external cooling for the vessel.

Period 6: Pressure Vessel Meltthrough to Collection as a Heap at Bottom of the Containment

As indicated, pressure vessel meltthrough might be expected in 30 to 120 min after the pipe rupture. Most of the core, with the melted portion of the vessel bottom, would break through very quickly. A major question concerns the interaction of the molten stream of metal with a water pool, which could be found at the bottom of the containment vessel as a result of ECCS operation during the LOCA.

The molten core material could cause steam explosions if it falls into a water pool. The predictive capability for estimating the probability and effects of steam explosions is very limited (Ref. 3-3). About all that can be stated with confidence is that explosions yielding localized pressures on the order of 3000 psi (200 atm) might occur.

The direct damaging effects of localized explosions in the sump area of the containment building are probably minor, because of the size and strength of the structures in the lower regions of these massive structures. The most serious effect of such explosions might be the overall incremental containment pressure associated with the explosions, and the tendency to scatter molten core materials to high sections of the

containment walls, where localized structural damage, perhaps even meltthrough, might occur.

Another serious effect of such localized explosions could be the promotion of subdivision of the molten core material with a resultant increase in the overall rate of the reaction of water with the metallic constituents of the molten core material. The resulting increased production of non-condensable hydrogen could be a particularly serious feature. It might lead to containment failure by overpressurization, in spite of successful operation of containment sprays or suppression pools, which are designed to reduce containment pressures by condensation of steam and other condensable fission products.

Period 7: Behavior of Molten Core Mass at the Bottom of the Containment

Upon failure of the pressure vessel bottom, it is estimated that as much as 45 to 75 percent of the core would fall into the containment vessel and that the self-heating of this material (after a total elapsed time of from 1 to 2 hr) would be of the order of 0.5 to 0.6 percent of the rated reactor power, or 15 to 20 MW (assuming that about 70 percent of the total decay heat is associated with the molten core materials).

The core material, with the molten vessel material, grid plate, etc., would weigh between 100 and 300 tons depending upon the mechanics of the melting process. The combined melted material would displace any water into which it fell and commence to heat the concrete, steel, etc., of the foundation base mat of the containment vessel. The rapid boiling of any water into which the molten core fell might scatter the molten core. The precise nature of the molten core and its flow behavior are uncertain, as discussed previously. Whether the core would rapidly spread out into a thin layer, remain more in a heap, or be broadcast by steam explosions is uncertain.

However, if one assumed the collection of the material on the floor as either a thin layer or heap, the concrete would spall and disintegrate as it was heated by the molten core. Since the molten fuel-steel mix is heavier than the concrete ingredients, the concrete residue could either mix with the core or float on the top. Currently preferred models for the core-concrete interaction postulate a turbulent interaction which thoroughly mixes and agitates the combined materials, resulting in expulsion of semi-volatile fission product isotopes from the molten material (Ref. 3-1).

Ordinary dry concrete has a density of about 145 to 155 lb/ft³ (specific gravity of 2.32 to 2.48) and contains about 2 percent by weight of water chemically bound to its molecular structure. Its thermal conductivity is about 1.2 to 2.0 Btu/hr ft^{°F} (or 0.005 to 0.008 cal/sec cm^{°C}) and its specific heat is about 0.20 to 0.24 Btu/lb^{°F} (or cal/g^{°C}), which are comparable to the thermal conductivity and specific heat of the melt. The concrete virtually disintegrates at about 932^{°F} (500^{°C}).

If there is an average thickness of 10 ft of concrete between the pressure vessel and the bottom of the foundation for the containment vessel, disintegration of from 100 to 400 tons of concrete might be required for the molten fuel to breach the containment vessel and enter the soil or rock beneath the structure.

It is estimated that the molten core might reach and penetrate the foundation of the containment structure in one to a few days after the beginning of the accident sequence.

If there is no water at the bottom of the containment, or if the water is boiled away from that region and condensed elsewhere, then the dry mass of molten core and concrete would be expected to vaporize at a significant rate. Recondensation of vaporized fuel and fission products would occur throughout the volume of the containment.

The lower three sketches in Figure 3-1 illustrate the three possible modes discussed above by which the containment structure might be breached and radioactivity released into the biosphere. The first mechanism depicted is a steam explosion which propels a portion of the pressure vessel forcefully enough to pierce the containment. The second mechanism which would occur at a later time (if it occurred at all) might result if the molten core melted through the pressure vessel and dropped to the concrete floor of the containment. As discussed, the steam and carbon dioxide produced in the interaction of the molten core with the concrete could pressurize the containment beyond its design limits. If one analyzes this a step further, the inadvisability of siting nuclear power reactors in limestone formations is apparent, wherein prolific and continual generation of carbon dioxide and steam would certainly exacerbate the danger of overpressurization. Finally, if the containment survived overpressurization, it would be breached eventually if the core melted through the concrete foundation into the soil or rock below. The arrows suggest the path by which the radioactivity in the molten core would penetrate the biosphere in the event of containment meltthrough. If the plant were located on the surface, the migration distance would be slightly greater than the radius of the containment; if the plant were underground, this distance would be increased by the burial depth. Groundwater contamination might be a problem which both underground and surface siting options would have in common in the event of meltthrough of the core in a Class-9 accident.

Table 3-1 provides a description of the mode of release of radioactive fission products from surface and underground plants due to the three modes of containment failure described here. For completeness, leakage through penetrations in the containment has been included as a failure mechanism. The possibility of breach of containment by projectiles launched by steam explosions is expected to be substantially reduced for underground sites to the point where it may become a negligible problem.

Table 3-1. Comparison of Potential Release Mechanisms in a Meltdown Accident

<u>Mode of Release</u>	<u>Surface Site</u>	<u>Underground Site</u>
Containment Wall Failure by Explosion/Missile Impact	Puff release to atmosphere	Possible seepage (puff release only if penetration closures fail)
Containment Failure by Overpressure	Puff release to atmosphere	Venting to atmosphere
Containment Failure by Meltthrough	Seepage through ~50 to 100 ft of rock/soil Groundwater contamination possible	Seepage through ~200 to 400 ft of rock/soil Groundwater contamination possible
Leakage Through Penetrations	Slow release to atmosphere	Slow release to atmosphere

However, failure by overpressure is expected to be a more serious problem. Figure 3-2 illustrates containment pressure-time histories derived by several investigators for accident sequences in which the ECCS was assumed to have failed. Although the conditions in each case differed in terms of reactor size and containment volume, one feature is common to all the results. In the first three to six hours, the pressure and temperature of the containment would increase steadily, leading to pressures greater than containment design limits within a matter of hours. The basic problem is that, in the absence of an adequate heat sink, heat is released to the containment atmosphere by fission product decay faster than it can be removed by conduction into the containment walls.¹ The net effect has been likened to a pressure cooker where the pressure increases continuously until it is relieved by containment failure.

3.1.2 Containment Pressure Management Methods

It is obvious that technology must be provided to control overpressurization of, and temperature buildup within the containment, if a Class-9 accident is to be dealt with in either an underground or surface facility. Several concepts have been considered which could provide required containment pressure management. The use of one (or some combination) of these may be required to remove overpressurization from the list of potential failure modes.

Heat sinks, such as suppression pools, pervious concrete, and external gravel bed expansion volumes, are passive and require no direct human activation or intervention. These concepts could be used to aid in condensation of steam and could supplement the heat transfer capability of the containment walls to prevent (or delay) overpressurization.

¹ The fission product decay heat rate is approximately proportional to $t^{-1/4}$, whereas the rate of heat transfer to the walls by conduction is approximately proportional to $t^{-1/2}$, where t is elapsed time.

NOTE: TYPICAL SURFACE CONTAINMENTS DESIGNED FOR 50-60 psig

Figure 3-2. Containment Pressure-Time Histories (ECCS Inoperable)

Conventional surface-sited BWR's are now provided with suppression pools; a similar (but perhaps larger) pool could be provided underground for any LWR. An outer containment liner of pervious concrete could also provide a large mass and internal surface for heat transfer and steam condensation, and may be required in any event to permit collection and drainage of groundwater seepage. The equivalent of this pervious concrete liner for a berm-sited plant could be provided by properly selected backfill, if a mechanism were provided for controlled release of gaseous and volatile materials out of the containment and into the surrounding backfill.

Pressure management could also be achieved by means of expansion volumes to supplement the containment structure volume. These could be provided separately in an underground installation although, with proper design, the existing tunnels and galleries of a mined-rock facility could be employed as readily available supplementary volumes.

As a backup safety measure, provision for specifically engineered, controlled, filtered venting systems would permit a release of excess pressure without a concurrent release of the more hazardous fission products. Such a release, although preferably avoided, nevertheless would be much less serious in its consequences than uncontrolled venting.

Very deep burial might appear to have the prospect of providing any desired capability for fission-product containment against overpressurization; however, by itself, such an approach does not appear promising. Experience from various fields of engineering involved with the storage, extraction, and transport of subsurface gases and liquids has been concisely summarized in Figure 3-3. Empirical results have indicated that leakage of gases may occur when pressures in rock tunnels exceed hydrostatic pressures (about 0.5 psi/ft of depth). At internal gas pressures in excess of the lithostatic pressure (equivalent to about

Figure 3-3. Containment Capability of Rock (Soil)

1 psi/ft of depth), leakage from unlined rock tunnels would be almost certain. Since the integrity of the containment boundary lining cannot be guaranteed under conditions of the postulated Class-9 accident, it must be assumed that gases within the containment would have direct access to the surrounding rock. It, therefore, appears prudent to assume that the overburden would provide an absolute containment capability of no more than about 0.5 psi/ft of depth.

Calculations have not been conducted to the point where pressure-time histories have reached pressure and thermal equilibrium conditions, since structural design pressures are customarily exceeded long before equilibrium occurs in surface facilities. In view of the calculations of Figure 3-2, which indicate pressures in the vicinity of 200 psi and rising after just a few hours, it appears that very large containment depths would be required, if equilibrium pressures were to be accommodated in an underground structure. Because all containment accessories, such as piping, seals, valves, etc., must also withstand these pressures to preserve containment integrity, the "brute force" method of going deeper into rock does not appear to offer a satisfactory solution.

The molten core is expected to melt through a typical 10 ft concrete foundation mat in perhaps 18 to 36 hr and then into the underlying soil or rock. Despite the uncertainties in the geometric configuration of the core mass, and its physical and chemical interactions with the concrete and rock beneath, consideration of heat transfer and decay heat rates suggests that total downward migration of the core will be of the order of 100 to 200 ft, at which time (perhaps months after core melt) it would probably be trapped in a slowly cooling vitreous slag ball (Ref. 3-7).

Because of the uncertain behavior of the molten core in the geological medium, it would be desirable to maintain some degree of control over the core melt displacement and the attendant fission products.

To this end, core-catchers of various configurations have been proposed. If some type of core-catcher were required for underground installations, it seems clear that the preferred configuration should be passive and should provide means for trapping the core decay heat in preference to permitting its escape into the containment atmosphere, with the attendant risk of venting. (Note that a core-catcher could serve a like purpose in a surface plant.)

3.1.3 Comparative Estimates of Underground and Surface Accident Consequences

If the containment should be breached, the impact of the released radioactivity on public health and safety has been analyzed for the important transport mechanisms of radionuclides in rock, soils, groundwater, and the atmosphere. The principal radionuclide transport mechanisms are illustrated in Figure 3-4. For a plant sited on the surface, a breach of the containment structure would result in a direct atmospheric release. The same breach in the case of an underground plant would release radioactivity to the rock or soil, and then into the atmosphere by ionic and gaseous dispersion through the medium, unless venting through natural, or manmade, fissures should occur. In each case, migration of radionuclides through soils and rock to underground aquifers must also be considered.

Transport of radionuclides by dispersion through the rock/soil medium is a slow process for essentially all fission product categories. However, other important potential escape paths into the atmosphere include leakage through penetrations, e.g., access tunnels and air-conditioning vents, and fissures (either naturally occurring or accident-induced) in the soil or rock overburden could provide pathways for more direct and rapid transport to the surface. Because these features would be strongly design and site-dependent, calculations have not yet been performed to evaluate transport by these mechanisms; and special

ATMOSPHERIC RELEASE AND SUBSEQUENT DISPERSION

Figure 3-4. Principal Radionuclide Transport Mechanisms .

methods would have to be devised to approximate (or bound) their impact. These methods will be identified and evaluated in the Detailed Study phase of the program.

Radionuclides may be considered in two categories, gaseous and ionic. The latter are those that enter into solution and interact chemically with the available groundwater and earth media into which they are released. Values of the parameters which govern the transport of radionuclides span a considerable range. For example, diffusion and dispersion coefficients for radionuclides in soils span 3 orders of magnitude. In dry soils, the ionic species are essentially immobile, whereas the gaseous species (e.g., noble gases, methyl iodide) will diffuse outward from the source toward regions of lower concentration. In general, this process is so slow that radioactive iodine, for example, will have decayed into insignificance for most practical underground burial depths long before the iodine gas has diffused to the surface.

Retention factors are measures of the speed of flow of ionic species relative to the groundwater speed. They also range over 3 orders of magnitude. For moist or water-saturated soils, the ionic species will dissolve in proportion to their ionic exchange potentials and will migrate by diffusion and fluid flow processes. The peak of the concentration will migrate in the direction of the water flow at some reduced speed defined by the retention factor. The ionic radionuclides will diffuse away from the concentration peak in accordance with dispersion laws.

The important geological parameters include the groundwater flow rate, which can vary over 5 orders of magnitude, and the soil bulk permeability, which can span 8 orders of magnitude. The latter is extremely difficult to quantify because of the potential non-uniformity of the geologic medium. In fact, determination of the bulk permeability may turn out to be one of the key problem areas in identifying and assessing the worth of potential underground sites.

Table 3-2 displays some of the important soil transport parameters for 54 radionuclides present at the time of containment failure. The parameters tabulated are representative values for transport in silty, fine sand. At an actual underground soil or rock site, it is anticipated that the design parameter values would be at least a factor of 10 smaller, i.e., the Table 3-2 values are representative of "worst-case" conditions. An examination of the tabulated parameters shows that two radionuclides, krypton-85 and strontium-90, appear to be the most troublesome to contain for reasonable burial depths. Krypton-85 is evidently the most difficult radionuclide to contain, from the point of view of gaseous diffusion, because of its long half-life (11 years), large source strength (0.56 E6 curies), and large dispersion coefficient. The thickness of containment required to prevent unacceptable exposure from krypton release should ensure negligible exposure from other gaseous radionuclides. This is not strictly true for soil depths up to 10 m where the radioactive release of xenon-133 evidently is as important as krypton-85 from the standpoint of biological implications. However, for deeper soil depths, the shorter half-life of xenon-133 will permit krypton release to become dominant. The radioactive iodine gaseous release is predicted to be vanishingly small for the burial depths considered here based on half-life considerations.

For the case of ionic radionuclide migration, the problem radioisotope appears to be strontium-90. This observation is based on a combination of its relatively long half-life (30 years), source strength (3.7 E6 curies) and reasonably large retention factor. At soil thicknesses sufficient to entrain strontium-90 until it has decayed to acceptable background levels, all other ionic species appear to be even more effectively entrained. For comparative purposes, cesium-137, which is produced in similar quantities to strontium-90 and has nearly the same half-life, is predicted to contribute less than one-tenth of the strontium-90 background radiation after transport through 10 m of soil thickness, because

Table 3-2. Initial Activity of Radionuclides in the Nuclear Reactor Core at the Time of the Hypothetical Accident

No.	Radionuclide	Strength, E8 curies	Half-Life, days	Dispersion ^a Coefficient, cm ² /s	Soil Retention Factor
1	Cobalt-58	0.0078	71.0	2 E-5	E-2
2	Cobalt-60	0.0029	1,920	2 E-5	E-2
3	Krypton-85	0.0056	3,950	0.015	N/A
4	Krypton-85m	0.24	0.183	0.015	N/A
5	Krypton-87	0.27	0.0528	0.015	N/A
6	Krypton-88	0.068	0.117	0.015	N/A
7	Rubidium-86	0.00026	18.7	2 E-5	E-2
8	Strontium-89	0.94	52.1	↑ ↓	2 E-3
9	Strontium-90	0.037	11,030		2 E-3
10	Strontium-91	1.1	0.403		2 E-3
11	Yttrium-90	0.039	2.67		2 E-4
12	Yttrium-91	1.2	59.0		2 E-4
13	Zirconium-95	1.5	65.2		1 E-2
14	Zirconium-97	1.5	0.71		↑ ↓
15	Niobium-95	1.5	35.0		
16	Molybdenum-99	1.6	2.8		
17	Technetium-99m	1.4	0.25		
18	Ruthenium-103	1.1	39.5		
19	Ruthenium-105	0.72	0.185		
20	Ruthenium-106	0.26	366		
21	Rhodium-105	0.49	1.50		
22	Tellurium-127	0.059	0.391		
23	Tellurium-127m	0.011	109		
24	Tellurium-129	0.31	0.048		
25	Tellurium-129m	0.053	0.340		
26	Tellurium-131m	0.13	1.25	2 E-5	1 E-2
27	Tellurium-132	1.2	3.25	2 E-5	1 E-2

$$^a D = G(D_M, 0.1 v_w)$$

Table 3-2. Initial Activity of Radionuclides in the Nuclear Reactor Core at the Time of the Hypothetical Accident (Cont'd)

No.	Radionuclide	Strength, E8 curies	Half-Life, days	Dispersion ^a Coefficient, cm ² /s	Soil Retention Factor
28	Antimony-127	0.061	3.88	2 E-5	1 E-2
29	Antimony-129	0.33	0.179	↑	1 E-2
30	Iodine-131	0.85	8.05	↑	N/A
31	Iodine-132	1.2	0.0958	↑	N/A
32	Iodine-133	1.7	0.875	↑	N/A
33	Iodine-134	1.9	0.0366	↓	N/A
34	Iodine-135	1.5	0.280	2 E-5	N/A
35	Xenon-133	1.7	5.28	0.015	N/A
36	Xenon-135	0.34	0.384	0.015	N/A
37	Cesium-134	0.075	750	2 E-5	2 E-4
38	Cesium-136	0.030	13.0	↑	2 E-4
39	Cesium-137	0.047	11,000	↑	2 E-4
40	Barium-140	1.6	12.8	↑	2 E-3
41	Lanthanum-140	1.6	1.67	↑	2 E-4
42	Cerium-141	1.5	32.3	↑	↑
43	Cerium-143	1.3	1.38	↑	↑
44	Cerium-144	0.85	284	↑	↓
45	Praseodymium-143	1.3	13.7	↑	2 E-4
46	Neodymium-147	0.60	11.1	↑	2 E-5
47	Neptunium-239	16.4	2.35	↑	2 E-5
48	Plutonium-238	0.00057	32,500	↑	↑
49	Plutonium-239	0.00021	8.9 E6	↑	↑
50	Plutonium-240	0.00021	2.4 E6	↑	↑
51	Plutonium-241	0.034	5,350	↑	↑
52	Americium-241	00.000017	1.5 E5	↑	↑
53	Curium-242	0.0050	163	↓	↓
54	Curium-244	0.00023	6,630	2 E-5	2 E-5

^a $D = G(D_M, 0.1v_w)$

its smaller retention factor reduces the propagation velocity by about a factor of 10.

A ground transport calculation has been performed in order to illustrate the analytical methods employed to bound the magnitude of the release, and to point out shortcomings in the approach. In this example, only the transport of krypton and strontium has been considered. The equations used for soil transport of the other radionuclide species are identical; only the input parameters would be changed.

Putting aside the complexities associated with the transport phenomena and the description of the media, the transport of radionuclides is rigorously described by the Boltzmann equation. In its simplest form, the Boltzmann equation equates the time rate of change of the radionuclide concentration to the radionuclide leakage. In symbols,

$$\frac{\partial C}{\partial t} = D \nabla^2 C$$

where

D = radionuclide diffusion coefficient

C = radionuclide concentration

The solution to this equation which satisfies the boundary conditions:

$$C \rightarrow 0 \quad \text{as} \quad t \rightarrow \infty$$

$$C \rightarrow 0 \quad \text{as} \quad t \rightarrow 0 \quad (x, y, z \text{ not all zero})$$

is:

$$C(x, y, z, t) = \frac{Q}{(4\pi Dt)^{3/2}} e^{-(x^2 + y^2 + z^2)/4Dt}$$

where x , y , and z are distances from the source along the three coordinate axis, t is the time, and

$$\int_{-\infty}^{\infty} \int_{-\infty}^{\infty} \int_{-\infty}^{\infty} C(x, y, z, t) \, dx \, dy \, dz = Q$$

is the total radionuclide inventory.

For the moment, radionuclide decay has been ignored. Radioactive decay reduces the spatial radionuclide concentration by the factor $e^{-\lambda t}$, where λ is related to the radionuclide half-life by $= \frac{\ln 2}{T^{1/2}}$

The current density of particle migration in any direction is given by Fick's Law of Diffusion

$$J = -D \nabla C$$

For the case of particles migrating to the surface of the earth (z is measured upward):

$$J_z = -D \nabla_z C = \frac{z}{2t} C$$

and the total current crossing the plane (x, y) located a distance z from the source is:

$$I_z = \int_{-\infty}^{\infty} \int_{-\infty}^{\infty} J_z \, dx \, dy = \frac{z}{2t} \frac{Q}{(4 \pi Dt)^{1/2}} e^{-z^2/4Dt}$$

The number of particles flowing in the positive z -direction is half the total number of particles:

$$\int_0^{\infty} I_z \, dt = Q/2$$

i.e., half the source diffuses through a plane located anywhere along the z -axis.

These equations were modified to account for particle removal as a consequence of radioactive decay. The concentration used in the calculation was:

$$C(x, y, z, t) = \frac{Q e^{-\lambda t}}{(4 \pi Dt)^{3/2}} e^{-(x^2 + y^2 + z^2)/4Dt}$$

and the total current through a plane at z was taken to be:

$$I_z = \frac{z}{2t} \frac{Qe^{-\lambda t}}{(4\pi Dt)^{1/2}} e^{-z^2/4Dt}$$

The total integrated release was calculated as

$$\int_0^\infty I_z(t) dt$$

For the case of a radionuclide release in a moving stream of water, z is replaced in the above equations by $(z - vt)$

where

$$v = R_f v_{\text{water}} = \text{radionuclide velocity}$$

and

$$R_f = \text{soil retention factor}$$

$$D = \text{greater of } D_M, 0.1 v_{\text{water}} \text{ (Ref. 3-8)}$$

where

$$D_M = \text{molecular dispersion coefficient.}$$

Except for the effect of radioactive decay, these equations also satisfy the principle of conservation of particles.

The dose rate to an individual positioned at a distance (x, y, z) from the point source is given by the product:

$$\text{Dose Rate} = \frac{1}{2} C(x, y, z, t) LK$$

where L is the mean free path of the emitted radiation in soil, and K is a conversion factor relating radionuclide decay rate to dose rate.

For the case of krypton-85, the concentration in the soil must be reduced by the branching ratio. That is to say 0.43 percent of the krypton-85 decays by emitting a 0.514 MeV gamma ray which contributes to the dose rate, whereas 99.6 percent of the krypton-85 decays by electron emission which would be absorbed by a very small thickness of soil.

Contributions to whole-body dose rate from radionuclides released to the atmosphere were not calculated.

The dose rate maxima were determined from the dose rate equation and were used to define the hazard zones of the figures following. In the example, results have been calculated for transport of radioactivity from a point source of radionuclides (krypton-85 and strontium-90) through the ground medium by dispersion and water flow. The soil was assumed to be a uniform, particulate, silty sand containing no cracks or fissures. In an actual case, the soil composition may vary on a macroscopic scale; i. e., there may be layers of different types of soil, and some cracks or fissures may be present. These latter, more realistic, cases can be modeled and calculated by means of large computers. They do not readily lend themselves to hand analysis.

The results of the calculations described in previous paragraphs are presented in Figure 3-5. The radioactive release rates are depicted for a planar surface 10 m from the point source. The vertical and the horizontal scales in the figure are logarithmic. It is seen that a significant fraction of the krypton inventory is released, whereas the strontium released at the surface is attenuated by a factor of nearly 1000, when compared to its initial release of radioactivity in the containment. This is understandable in terms of the speed of transport of the two nuclides. The peak of the krypton-85 release rate occurs at about 0.5 year, a small fraction of its 10.7 year half-life, while the peak of the strontium-90 release rate occurs at about 100 years, after several of its half-lives. The combination of the delayed release and radioactive decay is responsible for the marked attenuations calculated.

A similar calculation has been carried out for radioactive sources buried 10 and 30 m below the surface of the earth. The associated migration of the radionuclides to the surface of the earth has been calculated, and a hazard zone has been defined in terms of the biological consequences associated with their release. Figure 3-6 illustrates the results of such a calculation for the strontium-90 inventory following a containment failure.

Figure 3-5. Results of Soil/Rock Transport Calculation

Figure 3-6. Biological Consequences of Strontium Migration to Surface

The strontium migrates to the surface where it may be absorbed into the plant and animal life. The hazard zone is defined as that area in which the peak of the strontium-90 activity exceeds 2 microcuries/m², the maximum value assumed allowable (Ref. 3-1). Figure 3-6 shows that for a burial depth of 10 m, the radius of the hazard zone is 15 m. By the time the burial depth has reached 30 m, the radius of the hazard zone has shrunk to zero.

The results as presented should not be taken too literally because of the nature of the assumptions required to perform this simplified calculation. Actual hazard zones may be somewhat increased when the entire radionuclide inventory is considered, when actual soil parameters are used instead of idealized ones, when the effects of soil fissures and construction interfaces are included, or when a distributed source of radioactivity under extreme pressure is modeled rather than a simplified point source. The trend towards greatly reduced public exposure from undergrounding the reactor (as compared to surface siting) is clearly demonstrated, however.

A further objective was the comparison of atmospheric release rates and resulting hazard zones from surface facilities to those from underground plants. Two soil thicknesses were considered.

Because the soil effectively entrains all ionic species, only atmospheric release rates for krypton-85 have been presented. Figure 3-7 shows the results of these calculations.

For a surface plant, containment failure is taken as the instantaneous emission of the bulk of the fission product inventory from a serious Class-9 accident (Ref. 3-6). The case of the underground plant, on the other hand, with 10 or 30 m of overburden, shows that the effect of soil entrainment is reduction of the total atmospheric release of radioactivity by a factor of roughly 10⁴. Furthermore, a significant time delay is introduced before sensible radiation appears at the surface, and the total release is seen to be extended over a period of years. This suggests that, in the unlikely event

Figure 3-7. Atmospheric Release Rates

FIGURE 3-7

of a Class-9 accident, more time would be available for remedial actions or evacuation of potential hazard zones with underground plants than with surface facilities. Figure 3-8 shows the hazard zones associated with the release of krypton-85 from underground plants buried to depths of 10 and 30 m. Here, hazard zone has been defined as that distance from the reactor epicenter beyond which the dose rate to a man standing on the surface will not exceed 1 rad/yr. Whereas the hazard zone for surface sites is based on meteorological transport of all radionuclides, the hazard zones for the two burial depths considered are dominated by the krypton still entrained in a surface layer of soil about one mean-free-path thick.

It is seen that undergrounding has reduced the radius of the hazard zone by a factor of 1000 from that associated with the surface release of the reactor's radioactive fission-product inventory. These reduction factors should be taken as indicative of the general benefits of undergrounding, because of the approximations made in performing the calculations.

Evidently most radionuclides will be contained by a few meters of soil or rock, unless the integrity of the medium is compromised by cracks, fissures, or other open pathways to the surface, such as might occur at earth-structure interfaces. If careful site selection cannot clearly establish the native bulk transport properties of the medium, it may be necessary either to seal apparent fissures by high-pressure grouting, or to excavate the site and backfill with material having well-defined and controlled properties. Even then, periodic testing may be required over the lifetime of the nuclear plant to check on the integrity of the overburden and leakage paths associated with tunnels, shafts and other manmade penetrations.

If, in spite of these precautions, the requisite confidence cannot be established for adequate reductions in radioactivity released in a Class-9 accident, a controlled venting and filtering system concept should be investigated. Although less attractive than complete containment,

*APS ESTIMATE OF HAZARD ZONE, SURFACE
RELEASE OF 1.3×10^9 Ci (all radionuclides)

Figure 3-8. Biological Consequences of Radionuclide Release

it may prove to be a more desirable option than permitting the containment pressure to rise unchecked until the containment and protective overburden could be breached.

3.2 SITE CHARACTERISTICS

One of the objectives for the preliminary design study of underground nuclear installations was the determination of the existence of technically reasonable siting areas as part of a demonstration of concept feasibility. Rather than locating specific sites, however, the program was limited to identifying regions where geological siting criteria could reasonably be expected to fulfill requirements. The siting issue was addressed by first reviewing the requirements for a typical surface site, followed by an evaluation of those requirements peculiar to the problems of underground siting. A brief description has been provided of topographical, geological, and climatological attributes of California pertinent to technically acceptable underground sites.

3.2.1 Surface Plant Site Requirements

The Nuclear Regulatory Commission (NRC) Code of Federal Regulations (CFR), 10 CFR Part 50 (Ref. 3-9), specifies that a nuclear plant shall be designed to withstand certain natural phenomena (e.g., earthquakes, tornadoes, hurricanes, floods, tsunamis, seiches) without loss of capability to perform its safety functions. In addition, Appendix A of 10 CFR Part 100 identifies site evaluation criteria, such as (a) characteristics of the reactor design and proposed operation, (b) population density and use characteristics of the site environs, and (c) physical characteristics of the site, including seismology, meteorology, geology, and hydrology. The objective of these regulations is to ensure the public safety against any consequences, not only from the normal operation of the plant but also from unlikely accidents. Consequently, sites for the construction of nuclear plants are sought where the physical environment is relatively benign. It is not always possible to find locations where all of the environmental attributes are uniformly benign. Each proposed site must be examined in detail and evaluated in the light of possible design improvements. For a generic evaluation such as that conducted for this study, only certain attributes need specific attention and evaluation. These are defined as part of the site requirements.

3.2.1.1 Geophysical and Socio-Physical Parameters

The Preliminary Safety Analysis Report (PSAR) consolidates all of the information relevant to the CFR criteria for nuclear surface plants, (Ref. 3-10) into a single document. Table 3-3 lists the data typically determined for PSAR's.

3.2.1.2 Logistics and Support Parameters

When all other criteria have been satisfied, the final criterion for evaluating the suitability of a site is economics. The plant proposed for a given site must be able to deliver energy to the customer in an economically competitive manner. Those factors that are typically included in the logistics and support category are (a) distance to load center, (b) distance to cooling water supplies, (c) availability of housing for construction workers and operating personnel, and (d) existence of adequate schools, hospitals, etc. Because of the relatively high cost of transmission line construction, the distance to the grid interconnection to which the power plant must deliver power will provide one of the more important economic impacts.

3.2.2 Undergrounding Plant Site-Peculiar Requirements

An important consideration for the construction of an underground nuclear plant is the characteristics of the soil or rock at the site. For berm concepts the soil should consist of properties conducive to excavation and hauling by heavy-duty, high-capacity earth removal equipment with a minimum of blasting and ripping. Materials that generally have such properties include loam/clay mixtures, "tight" earth which is free of rocks, clays with some moisture and sand/gravel mixtures. The geographic distribution of such soils seems to be sufficiently widespread so that technically adequate sites can be located in almost any locale which meets the requirements for surface plants. This fact permits considerable flexibility in the overall siting process. For bermed underground plants, the geologic properties of the site must be adequate at

Table 3-3

Outline of Data Supplied in the PSAR

Site and Adjacent Areas

Location and Access
Highways
Traffic Patterns and Density of Vehicles

Population

Population Radial Distribution
Population Distance Projection

Land Use

Recreation Areas
Agricultural Use
Dairy and Livestock Farming
Urban Development

Plot Plan

Grading and Profile Maps
Site Development Plan

Meteorology

Temperature, Precipitation, and Air Mass Data
Atmospheric Stability
Wind Observation Data
Site Diffusion Data

Hydrology

Storm Runoff
Sea Erosion

Oceanography

Currents, Waves, Bottom Conditions, Marine Biology
Water Temperature
Tsunami

Continued --

Table 3-3 (Continued)

Geology

Physiography
Stratigraphy
Structures
Aerial Subsidence
Foundation Soil Investigations

Seismology

Seismic History
Site Seismology (Fault Locations)
Offshore Faults

depth as well as at the surface. Fortunately, natural processes generally bring about improvements in the subsurface medium automatically. For example, soil materials tend to be more consolidated with depth, a feature which improves their capability to serve as a foundation.

Rock cavity sites with desirable characteristics should allow excavation of large free-standing cavities. Generally, such construction requires a site with a sufficiently large monolithic rock block, free from major fault zones to contain all of the excavations for the facility without approaching the block boundaries. A "soft rock," such as limestone or sandstone would generally be preferred over "hard rock" such as granite or gneiss for lower excavation costs; although use of a limestone formation could reduce the effectiveness of accident management systems. However, in California, because of its preponderance of granite formations (e.g., the Sierra Nevada batholyte), a greater number of technically adequate hard-rock sites are likely to be found than would be the case for soft-rock sites. The topography for underground sites can be rugged and hilly. In fact, canyons and steep slopes, generally features not desirable for surface plants, may represent preferred sites for mined-rock concepts.

Because of the need for more complete determination of geologic properties at depth for underground plants at both soil and rock sites, the difficulty of the survey, the time required, and the cost of the survey, will be higher than equivalent costs for surface site investigations. Of particular concern is the ability to estimate the cost of large-scale underground excavation projects with reasonable accuracy. Attempts to develop accurate cost estimates of underground projects are routinely made in the course of preparation of designs for underground projects. However, there is increasing evidence that constructors of large-scale projects are experiencing difficulties in completing their work within estimated budgets. Analysis of contract bid prices and completed costs for a number of large (multimillion dollars) excavation projects reported in the open literature, indicates that most of the

projects were underbid by large factors. Actual costs were higher than bids by factors of from 1.5 to 3. Detailed investigations of the causes of overruns were beyond the scope of the tasks in this initial portion of the ERCDC program. However, it was the opinion of a large construction firm¹ that inadequate geologic information was the major contributing factor. Perhaps the explanation is that the effort required for an adequate geologic survey tends to be underestimated for many large underground projects. This may lead to expensive oversights which might have been avoided by more extensive preliminary investigations.

The location of the water table is particularly important for underground installations. The difficulties in dewatering can substantially increase the cost of construction and increase the possibility of flooding. Therefore, in rock-mined installations it would be advisable to avoid high permeability limestone formations in combination with a high water table. For excavated berm sites, to lower construction costs, it would be prudent to seek locations where the water table is as low as possible with respect to the depth of the excavation.

3.2.3 Generalized Attributes of California

3.2.3.1 Topographic

The main topographic features of California are the Coast Range which parallels the coastline from the Oregon border to a point north of Los Angeles, the Cascade Range which merges with the Coast Range on the north, the Sierra Nevada, and the Tehachapi Mountains. The Cascades extend further east and form a region of rugged terrain. Many streams and rivers, among them the Sacramento River, originate and flow to the coast from the Cascades. The Sierra Nevada range extends in a north-south direction along the eastern border of California, separated from the Coast Range by the Great Central Valley. The southern extreme of the

¹Private Communication, Morrison-Knudsen Corp., Inc., 25 February 1977.

Central Valley is bounded by the Tehachapies, which curve southwesterly to meet the Coast Range above Los Angeles. The region south-east of the Tehachapies consists of desert lands of the basin and range province (Mojave Desert) extending to the Arizona border at the Colorado River, and the Nevada border further north.

The Coastal Ranges are composed of sedimentary deposits with occasional basement rock outcrops. The Sierras are basically granite, of batholytic origin. The Cascades are volcanic in nature and of more recent origin than the Sierras (Refs. 3-11, 3-12).

Geologically acceptable rock sites for underground plants conceivably could be found in the granite mass of the Sierras. The Sierra foothills would be an advantageous and accessible siting area, since builders would avoid the obvious difficulties associated with the rugged terrain of the higher altitudes. Sites for rock cavities might also be found in the rocks of the basin and range province of the Mojave Desert, although it is unlikely that the number of candidate sites would be as great as in the Sierras. Sites also might be found in the Cascades; however, the relatively recent origin of these rocks and their associated potential for instability demands caution in the site selection process in this area.

The Coastal Region south of Los Angeles contains rocks with an overlay of thin sedimentary formations. These rocks, many of "basement-like" quality and character, could also conceivably host underground installations.

In selection of sites for berm (soil) covered plants, the broad areas of both the Central Valley and the Mojave Desert could be considered.

3.2.3.2 Geologic

A preliminary assessment of the geological properties of California was made to identify those areas in which more thorough evaluations should be conducted. It was decided that such an assessment could be made with only a minimum of detailed information. Further, it was

judged that the data compiled in Bulletin 198 of the State of California Division of Mines and Geology would suffice for making this preliminary assessment (Ref. 3-13).

Bulletin 198 (B 198) identifies 10 geologic problems that should be addressed in assessing natural hazards to the public. These problems are listed in Table 3-4 below:

Table 3-4. Geologic Problems

1. Earthquake
2. Loss of Mineral Resources
3. Land Slides
4. Floods
5. Erosion Activity
6. Expansive Soils
7. Fault Displacement
8. Volcanic Hazards
9. Tsunami Hazards
10. Subsidence

For the conduct of the California preliminary site investigation, it was decided that data from B 198 would be used for all problem areas except the Loss of Mineral Resources item. The objective of the preliminary investigation would be the identification of regions within the state where technically reasonable sites were likely to be found. From Table 3-4, data for each geologic problem were evaluated, and the areas identified with each hazard were indicated on maps. The individual maps of hazard areas were superimposed to produce the composite map shown in Figure 3-9. The unshaded portions of the map represent possible siting areas remaining after application of the exclusion criteria. Based on research analyses of geological maps of California, these areas were further analyzed to define geological regions of interest for both rock and soil siting. These locations are shown in Figure 3-10.

Figure 3-9. Technically Acceptable Siting Areas
(Unshaded) Remaining After Application of Exclusion Criteria

GEOLOGICAL AREAS OF INTEREST

ROCK(R)

- 1 PENINSULAR
- 2 EAST MOJAVE
- 3 CENTRAL VALLEY
- 4 KLAMATH MOUNTAINS

SOIL(S)

- 1 CENTRAL VALLEY
- 2 TRANSVERSE RANGE
- 3 EAST MOJAVE
- 4 SACRAMENTO VALLEY

Figure 3-10. Map of California Showing Remaining Technically Acceptable Areas (Hatched) Remaining After Application of Rock and Soil Criteria

Four areas were defined, though two generic regions, (a) the Klamath Mountains and Peninsular areas (for rock), and (b) the Transverse Range and Sacramento Valley (for soils), may have marginal potential. The remaining areas of geological interest are the Central Valley and East Mojave for both rock and soil sites. Examination of soil and rock drilling log data within the Central Valley area led to the identification of at least one location with very favorable rock characteristics. Moreover, there is strong reason to believe that other rock sites with comparable rock qualities exist elsewhere in the area. These drilling logs subsequently provided the basis for the rock models utilized in the specifications for design of rock cavity sites for the Detailed Study (second-half) portion of the ERCDC program. A similar effort was conducted, with use of the soil logs of the Rancho Seco power plant south of Sacramento, to develop a generic specification for the baseline soil model.

3.2.3.3 Climatology

The climatology of California is extremely diverse and variable. Climatic charts of the U.S. prepared by the National Climatic Center have divided California into seven drainage areas shown in Figure 3-11. These drainage areas have been defined for convenience in administration of the weather data-gathering network and do not necessarily indicate the existence of common regional weather conditions. However, the general climatic characteristics within the Central Valley and East Mojave areas appear to be individually quite uniform. An examination of the Environmental Technical Application Center records of weather parameters averaged over long periods (10-50 years) supports this observation (Ref. 3-14). Therefore, it was tentatively concluded that two meteorological descriptions might suffice for these two principal areas which have been recommended for analysis in the Detailed Study phase of the program.

Figure 3-11. Climatology Map of California (Ref. 3-15)

3.2.4 Conclusions

The necessity to evaluate sites carefully will not be substantially different for either underground or surface nuclear plant sites. However, some additional site restrictions may be imposed by electing to construct a nuclear power plant underground. Preliminary investigations show that there are large areas in California of technically acceptable siting for either berm or mined-rock plants. The final identification of specific areas which could be considered acceptable will depend upon detailed geotechnical, demographic and environmental investigations. An extension of the investigation reported here, which will include more parameters of interest in order to attempt to evaluate other factors of technical acceptability such as seismicity, volcanism, flooding hazards, demography, etc., will be conducted during the second-half, Detailed Study, portion of the ERCDC contract.

3.3

PRELIMINARY SEISMIC INVESTIGATIONS

In California, because of the state's earthquake history, seismic design of nuclear power plants is critical and is a significant factor for over-all plant construction. Some investigators have suggested that subsurface structures would have less stringent design factors relative to seismic events than surface structures, and that this difference could contribute to structural design sufficiently to result in lowered costs. In this study, the issues of underground containment structure response to a seismic disturbance were investigated by three approaches: (a) analysis of the variation of structural response with depth in representative earth media, (b) evaluation of the application of surface structure design criteria and analysis methodology to the underground structure, and (c) reviewing of the available seismic design methodologies for application to underground structures for both berm and mined-rock cavity concepts. Included in this summary of the results is an assessment of existing seismic design methodologies and their relevance to mined-rock cavity and bermed construction designs. The main objective of this portion of the study was the development of preliminary underground seismic design criteria. Much of the basic work described in this section was performed by consultants to The Aerospace Corporation, the Applied Nucleonics Company, Inc. (ANCO) of Santa Monica, California (Ref. 3-20).

3.3.1 Seismic Design Environment

The discussion of the seismic environment contained in this section has been limited to the random, oscillatory earth and structure motion caused by a seismic event. Landslides, liquefaction, and other gross motions or instabilities of the earth have not been discussed.

It is believed that the most significant effects of a seismic event upon a buried structure are the gross, overall motions induced in the structure. This may be illustrated by the following simple calculations. The case of a shear wave propagating upwards in a medium with a shear-wave velocity (c_s) of 1000 ft/sec is assumed. It is further assumed that the

dominant frequency of the motion is 1 Hz. Thus, the principal wavelength (λ) is

$$\lambda = \frac{c_s}{f} = 1000 \text{ ft}$$

One thousand feet is quite long compared to the structural dimensions of interest (several hundred feet), so that differential motions across the span of the structure should not be large. For example, if the medium has a density (ρ) of 120 lb/ft³ and a peak particle velocity (v) of 50 in/sec (corresponding to an acceleration of slightly less than 1 g) then the peak shear stress (τ) would be

$$\tau = \rho c_s v = 108 \text{ psi}$$

which is low compared to typical material strengths of structures. By comparison, the corresponding average strain is less than 3×10^{-3} in/in over a quarter wavelength (350 ft).

Thus, the significant aspects of the seismic environment are largely reduced to the specification of the motions caused by a seismic event. Inasmuch as most of the seismic motion data is from the free field (measurements made away from structures and at reasonable distances from the causative fault), this provides the starting point of the discussion of this section. Also, most of the seismic free field data has been measured at the earth surface, whereas current interest pertains to motion at depths on the order of 100 to 1000 ft. The available data on seismically induced motions at depth have been analyzed herein by comparing motions at depth to those at the surface for the same event.

In specifying the seismic environment, the primary problem has been that of characterizing the potential for damage caused by the tremors induced by a quake. The practice in the nuclear power industry has been to describe the motion by a standardized response spectrum, normalized to a site-specific value defined for the peak ground acceleration at which the reactor can be shut down safely without a major release of radioactivity.

The investigations necessary to establish the peak ground acceleration at the site of a proposed surface nuclear power plant are contained in Reference 3-18. The standardized and normalized response spectra of Reference 3-19 may then be multiplied by the peak acceleration obtained by the methods of Reference 3-18 (for the specific site) to yield a response spectrum for design of the facility. The spectral shape of Reference 3-19 is site-independent but an alternative shape may be derived by a site-dependent analysis when approved by the NRC.

3.3.2 Experimental Free-Field Motion Variation With Depth

Tables 3-5 and 3-6 summarize the results of free-field seismic-motion measurements taken at depth (Reference 3-20). It may be noted that the majority of data was taken in Japan and that all the observed events were small earthquakes. In addition, most of the data is peak acceleration expressed as a reduction factor relative to the surface value. All of the data show a reduction in motion at depth relative to the surface motion.

An inference of the motion variation with depth may also be derived from data obtained at the Humboldt Bay nuclear power plant in California. On June 7, 1975, this facility was subjected to moderate shaking caused by the 5.5 magnitude Ferndale earthquake, 15 miles distant. The strong-motion accelerographs at the site recorded a peak of 0.35g at the free-field surface in the transverse horizontal direction.

Prior to the event, a soil investigation and a soil/structure interaction calculation had been performed. The interaction calculations were performed using the SHAKE/LUSH codes, which are well accepted by the NRC, as noted in Reference 3-21. The schematic of the calculational model and selected results of its application prior to the Ferndale earthquake are shown in Figure 3-12. The calculations indicated an average ratio of

$$\frac{\text{reactor caisson acceleration at a depth of 84 ft}}{\text{free field acceleration at the surface}} = 0.535$$

The measured value of this same ratio was 0.459. The calculations were

Table 3-5. Reduction Factors - Maximum Vertical and Horizontal Responses - Soil^b

Description	Depth, m	R_A^a	R_V^a	R_D^a	Excitation	References
Hard loam	5.4	1/4.7	1/2.8	1/2.2	30 weak earthquakes	Okamoto, pg 109 F. Omore
Alluvial ground	10.0	negligible	--	--	Weak earthquakes	Saita
	20.0	1/3 - 1/5	--	--	Weak earthquakes	
30 m silt stratum overlying hard stratum	37.0	1/2 - 1/4	--	--	Microtremors	Okamoto, pp. 110-111
Alluvial ground with depth of	15.0	1/1.3 (horizontal)	--	--	Lafayette, Calif.	Time traces courtesy of Dr. Tom McEvilly, Dept of Geology and Geophysics (UC Berkeley)
40 m overlying bedrock	40.0	1/1.1 (vertical)	--	--	earthquake 1/8/77;	
		1/1.7 (horizontal)	--	--	peak site surface response was	
		1/3 (vertical)	--	--	0.08 g	
Layers of top-soil, gravel, silty sand to below 350 m; s-wave velocity approximately 150 m/sec at surface, then 300 m/sec from -10 m to -70 m	50.0	1/2.5 (averaged)	--	--	Hachinohe site; peak surface accelerations up to 0.035; accelerometers at surface and -50 m; 10 records	Tanaka (1973)

^a R_A = Reduction Factor, Acceleration; R_V = Reduction Factor, Velocity; R_D = Reduction Factor, Displacement

^b This table is from "Seismic Assessment of Underground and Buried Nuclear Power Plants", Report #1226-1, Applied Nucleonics Co. Inc., Los Angeles, Calif., September 1977.

Table 3-5. Reduction Factors - Maximum Vertical and Horizontal Responses - Soil^b (Continued)

Description	Depth, m	R_A^a	R_V^a	R_D^a	Excitation	References
Layers of loam, sand, silt, clay, gravel to depth of 78 m overlying mudstone; s-wave velocities from 120 m/sec at -10 m to greater than 400 m/sec below -15 m	80.0	1/3 (averaged)	--	--	Tokyo University site; accelerometers at surface and -80 m (in stone); 10 earthquake records with 0.06 g to 0.10 g peak surface acceleration	Tanaka (1973)
Layers of silty sand and clay to depth of 24 m	7.0	1/2 (averaged)	--	--		
overlying gravelly sand/sandy clays to 244 m	15.0	1/2.7 (averaged)	--	--		
	24.0	1/4 (averaged)	--	--		

^a R_A = Reduction Factor, Acceleration; R_V = Reduction Factor, Velocity; R_D = Reduction Factor, Displacement

^b This table is from "Seismic Assessment of Underground and Buried Nuclear Power Plants", Report #1226-1, Applied Nucleonics Co. Inc., Los Angeles, Calif., September 1977.

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Table 3-6. Reduction Factors in Maximum Horizontal Response - Rock^a

Description	Depth, m	R_A	F_V	R_D	Excitation	Notes	References
Hard, coarse-grained tuff; p-wave velocity	67	1/1.5 - 1/2	--	1/1	4 small earthquakes 0.01 to 0.08 g peak surface	Kinugawa Power Station, Japan	Okamoto and Mizukoshi (1967)
Average 3700 m/sec	34	1/1 - 1/1.5	--	--	Acceleration		
Liparite	38	1/2	--	--	0.01 g surface motion	Chiba Power Station	Okamoto (1973)
Amphibolite	150	--	--	1/1.2 1/1.7	15 small earthquakes	Hitachi Mine displacements measurements at 0, 150, 300, 450 m	Kenai (1951)

^a This table is from "Seismic Assessment of Underground and Buried Nuclear Power Plants", Report #1226-1, Applied Nucleonics Co. Inc., Los Angeles, Calif., September 1977.

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Location	Elevation, ft	Max. Accel. for Two Postulated Surface Motions, g	
Free-field (storage building)	0	0.40	0.25
Refueling Building	0	0.23	0.15
Reactor Caisson	-84	0.22	0.13

Figure 3-12. Model and Results from Pre-Earthquake Calculations
(Humboldt Bay Nuclear Plant)

slightly conservative relative to the measurements, as would be desired. However, the agreement appears to be adequate.

The records obtained during the Ferndale earthquake are indicated in Figure 3-13. After the event, additional calculations were performed, as reported in Reference 3-22. Some of the results of these calculations, together with associated recorded quantities, are shown in Figure 3-14. This figure indicates that the measured and calculated structural response spectra agree quite well in amplitude and frequency content. The peak free field acceleration that was used in these calculations is also shown in Figure 3-14 and indicates a substantial attenuation with depth. Because the calculations of structure motion agreed well with measurements, it may be argued that the calculated free field motions are also reasonably accurate for the event. Thus, this event can be used as further evidence that free field accelerations decay with depth. This conclusion is important, because the surface acceleration here is much larger than for any other event in which motion variation with depth has been measured.

3.3.3 Calculated Free Field Motion Variation with Depth

It may be seen from the preceding section that the available experimental data on the variation of seismic motion with depth are very sparse. In order to provide additional guidance on the variation of motion with depth, a simplified analysis was performed by ANCO, as shown in Reference 3-20.

The starting point for these calculations was the generic site description of Table 3-7 with the assumed properties of low-strain compressional-wave velocity, density, and Poisson's ratio, as shown. From the low-strain compressional-wave velocity, density and Poisson's ratio, the low-strain shear modulus may be determined by one of the fundamental relationships of elasticity. This modulus was then reduced by ANCO as recommended in Reference 3-23, because of the large-strain amplitudes associated with the postulated design basis earthquakes of interest. The reduced shear modulus and the density can be combined to estimate the

Figure 3-13. Experimental Data Obtained at the Humboldt Bay Nuclear Power Plant During the Ferndale Earthquake

Figure 3-14. Computed and Recorded Response Quantities for the Humboldt Bay Nuclear Power Plant for the Ferndale Earthquake

Table 3-7. Material Properties Used in Calculation of Free Field
Motion Variation with Depth

	Depth, ft	Low-Strain Compressional Wave Velocity, ft/sec	Density, lb/ft ³	Poisson's Ratio	Low-Strain Shear Modulus, psi	High-Strain Shear Modulus, psi	High-Strain Shear Velocity, ft/sec	High-Strain Compressional Velocity, ft/sec
Hard Silt (Saturated)	100	2000	110	0.37	2 E4	.75 E4	562	1240
Silty Clay (Saturated)	200	4000	130	0.37	9.2 E4	5.5 E4	1400	3080
Dense Sands and Gravel (Saturated)	400	5000	135	0.33	1.8 E5	1.5 E5	2270	4500

high-strain shear and compressional-wave velocities shown in the last two columns of Table 3-7.

The properties discussed above were used by ANCO in two-dimensional, plane-strain, linear finite element calculations for which the mesh is shown schematically in Figure 3-15. The calculations were performed using a modal technique in which 30 modes were utilized and 10 percent critical damping was assumed in each mode. Actual earthquake time-histories, scaled to yield an NRC Regulatory Guide 1.60 spectrum of about 0.5g were applied in the vertical and horizontal directions at the base of the mesh.

The free field motion results obtained from this calculation are shown in Figure 3-16 and Figure 3-17. The response spectra of the input (bedrock) motion are shown, as well as response spectra for selected elevations in the soil profile. It may be noted that the response spectra for the surface are everywhere greater than the response spectra for bedrock.

3.3.4 Tentative Recommended Reduction Factors for Free Field Motion

Analysis of the calculated response spectra of Reference 3-20 led to the reduction factors given below for acceleration (R_A), velocity (R_V) and displacement (R_D).

Table 3-8. Calculated Free Field Motion Reduction Factors

Direction	Depth, m	R_A	R_V	R_D
Horizontal	-32	1/2.3	1/2.3	1/1.6
	-64	1/2.3	1/3.0	1/2.1
Vertical	-32	1/2.2	1/2.5	1/1.4
	-64	1/3.3	1/3.4	1/2.2

Consideration of both the calculated response spectra and the experimental data presented in Tables 3-5 and 3-6 led to more conservative proposed reduction factors for preliminary design, shown in Table 3-9.

Low-Strain
Compressional
Wave Velocity,
ft/s

Figure 3-15. Finite Element Model for Calculation of Freefield Motion Variation With Depth
(Reproduced from Ref. 3-20)

Figure 3-16. Horizontal Free Field Spectra Computed from Finite-Element Model
(Reproduced from Ref. 3-20)

Figure 3-17. Vertical Free Field Spectra Computed from Finite-Element Model
(Reproduced from Ref. 3-20)

Table 3-9. Proposed Reduction Factors for Preliminary Design
(Depth ≥ 30 m)

	R_A	R_V	R_D
Soil	1/2.0	1/1.5	1/1.2
Rock	1/1.5	1/1.2	1/1.0

The recommended reduction factors of Table 3-9 allow distinction between soil and rock sites and are independent of depth. At depths of less than 30 m, it is recommended that a linear transition be made between a unity reduction factor appropriate for the surface and the reduction factor of Table 3-9. That is $R_A(20)$ for a soil site at a depth of 20 m would be

$$\begin{aligned}
 R_A(y) &= 1 - \frac{y}{30} (1 - R_A) \\
 &= 1 - \frac{20}{30} (1 - \frac{1}{2}) = \frac{2}{3}
 \end{aligned}$$

where y is the depth in meters.

The method of application of these free field motion reduction factors to structural design is explained in Section 3.3.5.

3.3.5 Calculated Structural Motion Variation with Depth

Calculations were performed by ANCO, and reported in Reference 3-20, on the variation of structural response with depth of burial of the structure. The calculations were performed with the aid of a two-dimensional, plane-strain, elastic finite-element code, using the mesh depicted schematically in Figure 3-18. The properties of the free field were the same as those discussed in Section 3.3.2.

The shaded area of Figure 3-18 is intended to indicate the location of a simulated structure of a surface nuclear power plant containment building as modeled in the calculations. The elastic properties selected to represent the gross mass and stiffness properties of a typical containment structure are given in Figure 3-19. They were developed from

Figure 3-18. Finite Element Model for Calculation of Structural Response Variation with Depth of Burial
(Reproduced from Ref. 3-20)

EQUIVALENT PLANE STRAIN
FINITE ELEMENT MODEL

TABULATION OF MATERIAL PROPERTIES OF MODEL

Relative Elevation m	ρ_e kg/m ³	G_e N/m ² (xE-10)	E_e N/m ² (xE-10)	μ
28 to 32	425	0.24	0.55	0.15
24 to 28	550	0.28	0.64	0.15
20 to 21	675	0.32	0.74	0.15
16 to 20	800	0.36	0.83	0.15
12 to 16	925	0.40	0.92	0.15
8 to 12	1050	0.45	1.04	0.15
4 to 8	1160	0.50	1.15	0.15
0 to 4	2400	1.10	2.5	0.15

PROPERTIES OF CONCRETE:

$$\rho = 2400 \text{ kg/m}^3$$

$$G = 1.1 \times 10^{10} \text{ N/m}^2$$

$$E = 2.5 \times 10^{10} \text{ N/m}^2$$

$$\mu = 0.15 \text{ (Poisson's Ratio)}$$

Figure 3-19. Mass and Stiffness Properties Chosen to Represent
Typical Containment Structure

analyses of several actual nuclear power plant structures. For the case of a buried structure the modeled containment building shown in the shaded area of Figure 3-18 was translated into an appropriate subsurface location where the corresponding elements of the calculational mesh were assigned the structural properties of Figure 3-19.

With the same computational scheme and excitation as discussed in Section 3.3.3, ANCO computed the horizontal and vertical roof and base response spectra for the following two buried plant configurations:

- Surface plant - as shown in Figure 3-18
- Berm plant - roof of structure flush with the free field surface

The results are shown in Figures 3-20 through 3-23. The data output point for the structure was taken at its edge in order to detect any rocking motion of the structure.

To add physical insight to the structural response data of Figures 3-20 through 3-23, the data were processed by the following method. The horizontal and vertical structural response spectra were divided by the associated horizontal and vertical free field surface spectra to derive structural influence factors. The results were then smoothed and expressed in analytical form as shown in Table 3-10. These factors are defined as

$$V_{\text{structure}} = K * V_{\text{surface free field}}$$

That is, the structural response spectra have been derived by appropriately modifying the surface free field spectra. The results show a suppression of the motion for the bermed structure as contrasted with the surface structure. Thus, the horizontal structural influence factor for the roof of the structure at 2 Hz is 0.8 for the berm case and 1.9 for the surface case. The surface case indicates significant horizontal-motion variation with elevation on the structure and some amplification due to the natural rocking frequency of the soil-structure system. These effects are diminished for the bermed case. The responses indicate a general diminution at high frequencies, which would be expected.

Figure 3-20. Horizontal Roof Spectra, Typical Central Valley Site
(Reproduced from Ref. 3-20)

Figure 3-21. Horizontal Base Spectra - Typical Central Valley Site
(Reproduced from Ref. 3-20)

Figure 3-22. Vertical Roof Spectra, Typical Central Valley Site
(Reproduced from Ref. 3-20)

Figure 3-23. Vertical Base Spectra - Typical Central Valley Site
(Reproduced from Ref. 3-20)

Table 3-10. Horizontal and Vertical Structural Influence Factors to be Applied to Associated Horizontal and Vertical Free Field Surface Response Spectra

Structure Location	Structural Influence Factor (K)	
	Horizontal	Vertical
Structure Founded on Surface	$1 - 0.2 \left(\frac{f-0.6}{1.4} \right) + \left[0.3 + 0.8 \left(\frac{f-0.6}{1.4} \right) \right] \frac{x}{h} ; 0.6 < f < 2$ $0.8 + 1.1 \frac{x}{h} ; 2 < f < 10$	$1.1 - 0.3 \left(\frac{f-0.6}{1.4} \right) ; 0.6 < f < 2$ $0.8 ; 2 < f < 10$
Top of Structure Flush with Surface	$0.6 + 0.1 + \left[0.1 \left(\frac{f-0.6}{1.4} \right) \right] \frac{x}{h} ; 0.6 < f < 2$ $0.6 + 0.2 \frac{x}{h} ; 2 < f < 10$	$1.1 - 0.4 \left(\frac{f-0.6}{1.4} \right) ; 0.6 < f < 2$ $0.7 ; 2 < f < 10$

Notes: h = Height of Structure
x = Position Above Base of Structure
f = Frequency Abcissa of Response Spectrum, Hz

The calculations performed by ANCO were not intended to be an exact representation of the motion of a containment structure in a strong seismic environment. It is pointed out in Reference 3-20 that substantial differences exist between linear-model calculations of structural response and non-linear-soil calculations of structural response for the same problem. Considerable progress has been made in the development of the technology for conducting non-linear analyses of soil-structure interactions. However, this technology requires advancement by means of further application and comparison with experimental results before it can be considered to be applicable for the preliminary seismic design of underground structures. However, it is believed that the ANCO calculations provide a valid estimate of the relative seismic benefits of structural burial.

3.3.6 Applicability of Current Methodologies for Estimation of Criteria and Analysis of Structures and Equipment

The method approved by the NRC for the specification of free field surface motion caused by an earthquake involves the specification of a peak acceleration for a specific site following the methods of investigation defined in Reference 3-18 with the adoption of the spectral shapes of Reference 3-19. The recommended spectra of Reference 3-19 for horizontal and vertical motion have been reproduced as Figures 3-24 and 3-25, respectively. Reference 3-19, as noted, permits the use of a site-dependent spectral shape rather than the models of Figures 3-24 and 3-25, if the site-dependent shape can be justified by analysis. The procedures of References 3-18 and 3-19 are judged to be adequate for the specification of surface motion for underground nuclear power plants.

The reduction factors proposed in Reference 3-20 and listed above in Table 3-9 are recommended for specification of free field response spectra at depth. The horizontal spectra obtained by applying these factors to the NRC spectra are shown in Figure 3-26.

Figure 3-24. Horizontal Design Response Spectra - Scaled to 1g Horizontal Ground Acceleration
(Reproduced from Ref. 3-20)

Figure 3-25. Vertical Design Response Spectra - Scaled to 1g Horizontal Ground Acceleration
(Reproduced from Ref. 3-20)

Figure 3-26. Reduced 0.5 g NRC Horizontal Response Spectra for Depths Greater than 30
(Reproduced from Ref. 3-20)

There is also an NRC regulatory precedent for permitting a reduction of the motion specification, with depth in accordance with methodology contained in an NRC Standard Review Plan. A Standard Review Plan (SRP) provides guidance to utilities and architect/engineers in preparing applications to construct nuclear power plants. In the SRP of Reference 3-21 the NRC staff has stated:

"Using an appropriate analysis method, with appropriate soil properties, obtain a time history at the base of the idealized soil profile. ---when the time history obtained is applied at the base of the idealized soil profile, using appropriate soil properties, the vibratory motion calculated at the elevation of Category I structural foundations should, in general, give response spectra at all frequencies (0.2 cps to 50 cps) not less than 60% of the design response spectra. ---Response spectral values in the idealized soil profile at the foundation level --- that are less than 60% of the corresponding design response spectral values may be acceptable, provided they can be justified. "

This portion of the plan states that it is NRC practice to permit a foundation level input of 60 percent of the surface motion, or less, provided the recommended attenuation can be justified. This is an important precedent contained in current nuclear power plant structural design practice, which may be applicable to future evaluations of licensing requirements for bermed and rock-sited nuclear power plants.

By implication from Reference 3-21, the NRC permits the use of appropriate soil-structure interaction codes, such as FLUSH and LUSH of References 3-24 and 3-25, as a basis of design for surface-founded and slightly embedded structures. It is anticipated that the use of such codes would be permitted also for the bermed and buried structures of interest in this study. The results shown earlier for the deeply embedded Humboldt Bay reactor caisson were obtained with use of the FLUSH code and the agreement between experimental and calculated results shown therein is

quite impressive. The structural influence factors of Table 3-10 may be used for guidance or preliminary estimates of structure motions. However, with the FLUSH and LUSH codes readily available, it appears that the necessary technology for adequate structure motion calculations for underground plants already exists.

It is anticipated that there would be no changes in the design methodology for equipment mounted within underground structures. These items would simply be designed in accordance with appropriately derived wall or floor response spectra.

3.3.7 Implications of Seismic Investigations

In summary, the results shown indicate rather substantial decreases in seismic motions with depth in both soft and hard earth media. Observations from both empirical data and numerical results indicate that accelerations should be reduced by at least a factor of 0.5 in soils, or a factor of 0.667 in rock, at depths of 100 ft or greater. These reductions in free field accelerations are apparently sufficient to also reduce structural motions substantially with depth. Reductions in input and structural motions of the amounts indicated may result in sufficiently less severe environments to permit corresponding reductions in the amounts of steel and concrete needed to resist dynamic forces in underground structures. The reduced motions may also permit the use of more standardized equipment within the underground facilities, i. e., equipment designed for the more benign seismic environments of the midwest and eastern portions of the U. S.

These reduction factors for free field accelerations offer implicit hope that some cost advantages could result from underground construction with respect to seismic design requirements. However, no specific investigations related to cost impacts of underground seismic construction were made for this study. Thus, no final conclusions can be made in this regard. It is expected that better cost information will be developed during the Detailed Study portion of the second-half of the ERCDC program, in which architect/engineers will develop designs and

make cost estimates for buried plants.

The results presented in Section 3.3 imply that current seismic structural design methods will probably be adequate for analyzing underground structures, as well as surface structures. Though an evident need exists to improve the physical basis for seismic analyses to incorporate more realistic, non-linear-soil stress-strain relationships, current analytical methods have had reasonably good success in post-incident "predictions" of the results of earthquake interactions with structures, in a limited number of cases. These results lend credibility to the conclusion that current seismic design methods are probably adequate for at least preliminary design of underground facilities.

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4.0 ANALYSIS OF DESIGN AND COST IMPACTS OF UNDERGROUNDING NUCLEAR POWER PLANTS

In this section, a review is provided of the factors which will result in practical and economical underground nuclear power plant construction. Building structures underground is not a new or novel concept. Many types of facilities, such as manufacturing plants, warehouses, personnel shelters, military headquarters, hydroelectric plants, etc., have been constructed underground for a variety of reasons. A major motivation for locating such facilities underground has been that of protection against military surface threats, such as bomb explosions. However, availability of adequate siting areas has also been a common motivation for selecting an underground construction mode. The design of subsurface structures for nuclear power plants produces a number of unique design requirements when the motivation is primarily the reduction of public exposure from accidentally released radioactivity.

Several approaches have been considered for the design of underground nuclear plants which might reduce radioactive exposure to the public from severe, accidentally-induced releases. As in any engineering undertaking, among the principal objectives of such designs are the maximization of performance and the minimization of costs. The following discussions present an analysis of a variety of different design configurations for both berm and rock cavity underground concepts with respect to these objectives: (a) exploring and analyzing the attributes of each configuration, (b) providing estimates of the elements of the plant for which major cost impacts will occur for underground facilities, and (c) providing a comparative evaluation of several berm and rock cavity underground concepts.

4.1 UNDERGROUND PLANT CONCEPTS AND GEOLOGIC CRITERIA

In this section, a description is given of several basic berm containment and mined-rock underground concepts together with a discussion of geologic design requirements for emplacing a nuclear power plant beneath the surface of the earth. As defined in the introduction, there are two fundamentally different methods by which a surface facility can be placed underground. The first is a conceptually simple burial method performed basically by covering a suitably strengthened surface plant with soil (berm containment). In the second method caverns are excavated in a large body of rock (mined-rock cavity) and the facility constructed within the excavation. There are several alternatives for implementing these two basic underground construction methods. The various concepts defined below will have different total costs because of differences in the details of the engineering and construction of the covering (berm plants), the lengths of equipment interconnections, and the method and difficulty of access from the surface during construction. These basic engineering issues are defined and discussed initially in this section without presenting detailed cost relationships. Estimates of related cost impacts of underground construction are given subsequently in Sections 4.3 through 4.5.

Whether or not a below-ground facility can be constructed in a practical and economic way is largely dependent on site geology. Discussions of the essential factors required for satisfactory sites from a geological point of view are given below. The discussion has been presented in terms of the important characteristics of soils and rocks that influence their performance as structural materials, as well as their capacity to provide a heat and fission product storage medium.

4.1.1 Berm Configuration

The most elementary concept for a berm-contained power plant consists of a facility covered with a layer of soil. Some additional analyses of how the soil cover is to be applied, the source from which the soil is to

be obtained, and the cost and disposition of the excavation products, suggest that several design variations for soil-covered plants are worthy of consideration. Three basic soil-covered concepts (cover-only, cut-and-cover, and hillside) are illustrated in the sketches of Figure 4-1. Additional variations on these three basic concepts might result from specific site or plant requirements and limitations.

The basic reason for using cut-and-cover construction rather than simply covering a surface-sited facility with soil is to provide a source of low-cost, easily accessible soil cover (fill) material for the berm plant. Cut-and-cover construction techniques provide a simple direct means for balancing fill production costs (of excavation) against the cost of the soil covering for the plant. The cut-and-cover configuration can also reduce the above ground profile by lowering the overall height of the power plant relative to the surface of the terrain thereby providing some aesthetic and psychological benefits for the construction method. If the intended location site is in a hilly area, "rolling" topography, or a secluded or hidden area, an above ground "cover-only" configuration might be altogether economically, technologically, and aesthetically acceptable. Thus, the topographical features of siting may be important factors with respect to selection of one of several concepts for the design of a berm-contained facility.

Geologic factors are also of major importance for the berm concepts. The dominant geological criteria that must be applied are those relating to soil properties at the site, depth of water table, and proximity to flood plains. Sites for both cut-and-cover and hillside configurations of the berm models should include soil to the level of the base of the foundations and a more consolidated (or rock) material beneath for the foundation. Although the soil and terrain descriptions presented in the subsequent discussion tend to be idealized from an engineering and construction standpoint, the conclusions are quite general. Soil inadequacies, if they exist, can be overcome at sites with less than ideal geology by various construction techniques. The economic penalty for such site

Figure 4-1. Berm Configuration Concepts

improvement procedures is estimated to be minor in magnitude relative to the overall cost of the project for most soil conditions.

4.1.1.1 Cut-and-Cover Concept

The fully-buried, cut-and-cover berm concept, shown schematically in Figure 4-2, incorporates a reactor containment building 140 ft in diameter by 275 ft high, constructed entirely below ground. The structure is a high-strength, relatively stiff, steel and concrete building capable of withstanding the loads imposed upon it by the soil cover. For simplification in the sketch, the nuclear reactor containment building is the only structure shown. The soil has been shown as excavated to provide a wall slope of 45 deg. The base of the pit is sized to provide ample working space around the exterior of the building during construction. In the illustration, 75 ft of annular space has been provided for construction, material handling and erection purposes. Thus, the pit is 325 ft deep, 290 ft wide at the base, with a width of 940 ft at the ground surface level. On completion of the excavation process, the containment structure would be constructed in much the same manner as an above ground facility. All interconnecting corridors, pipe enclosures, land-line conduits, and access shafts or tunnels would be installed prior to commencement of the back-filling procedures.

In this cut-and-cover configuration, the top of the fill material has been shown to be essentially level with the local ground surface. For a preliminary model it has been assumed that the depth of cover over the top of the reactor chamber is 50 ft. Thus, non-expansive soil is desirable at depths between approximately 275 and 350 ft. The surface material to a depth of approximately 15 ft should also be non-expansive soil. The intervening soils can be almost any type as long as there are permeable layers at depth which give adequate site drainage in the event of prolonged rainfall. The depth to the water table should be at least 300 ft to minimize problems from this source; however, in practice this condition could be difficult to satisfy.

Figure 4-2. Cut-and-Cover Configuration

For this configuration, moderately hilly or rolling topography could be advantageous, depending on the characteristics of the final facility design. To reduce flooding hazard, the facility should be sited beyond the boundaries of the 100-year storm flood plain. Normal concerns for soil liquefaction under earthquake loadings in the facility area should be ameliorated by designs of this type, due to the massive surcharge of the fill material.

The structure must be designed to withstand the lateral earth pressure on the walls and the pressure of the overburden on the roof. Unless a means is devised to transfer the roof loads into the surrounding earth, the walls will also have to be designed to support the vertical roof loads. The structure that must support the soil cover loads is estimated to be significantly more massive than the containment structures of surface sited plants. The types of structural shapes that would most efficiently support the soil cover loads are cylindrical or spherical. Substantial, internal cross-bracing may be required to help support the wall loads. If this requirement could be coupled with reinforced concrete lattice-type reactor and steam generator supports, some economies might be realizable. Design efforts to reduce the span and height of the reactor containment building would also result in very substantial savings. Finally, if a portion of the roof loads could be transmitted directly into the surrounding soil, the mass of the structure could also then be reduced.

The earthen backfill should be carefully selected for permeability characteristics. A lining for the sides of the excavation, and the upper 50 ft of cover, consisting of a highly impermeable clay, might be utilized to restrict lateral and upward migration of radioactive material that might escape from the reactor containment building in the case of a severe accident. The volume of material excavated from the configuration shown is approximately $4.6 \times 10^6 \text{ yd}^3$.

The ancillary support buildings, which may be undergrounded, could be placed either around the containment building, at the base of the pit (which would have to be enlarged to facilitate subgrade location), or at a higher level within the pit. If placed at the higher level, unique constraints

are imposed on construction scheduling because of the potential need for partial filling of the pit during the construction cycle. While this approach may appear to offer economy of construction, it could force serial scheduling of construction operations which might produce a lengthened construction cycle and associated increases in cost. Personnel access for maintenance, and operation (and possibly some late construction work) would probably be achieved through an inclined tunnel or by means of an elevator. Hence, there could also be a significant operational disadvantage in this regard.

A variation of the cut-and-cover concept using a site with near-surface rock was considered. This concept as shown in Figure 4-3, embodies a shallow-sited facility in a competent rock material, which would probably be covered by soil. As shown in the figure, all (or much of) the wall portion of the reactor containment building would be emplaced within the near-surface rock mass. The cavity would be constructed by removing any overburden that might exist above the competent rock surface. The cavity could be contoured to the facility dimensions by excavating and removing material utilizing quarrying, or bench mining techniques. The structure could then be constructed within the cavity with a structurally adequate roof spanning the rock openings. The excavated volume over the roof has been assumed to be backfilled with suitable materials, as in the concepts discussed previously.

For this type of construction, a layer of soil overburden of variable thickness would normally be expected above the natural rock surface. However, the upper portion of the rock mass would probably have undergone some weathering and alteration with a resulting decrease in the quality of its structural properties. The thickness of the weathered zone could vary from a few feet to depths approaching 100 ft depending on the type of rock and the aging mechanisms. Past experience in subsurface siting exploration indicates that a weathered region of less than 50 ft in thickness may be reasonably anticipated if such a consideration is included in the site survey and selection criteria. Thus, it is conceivable that the walls of the facility would have nominal construction requirements in the

Figure 4-3. Cut-and-Cover Variation - Partial Rock Formation

region below the weathered zone, while a more substantial structure might be required to maintain wall integrity through the weathered zone. For conceptual design purposes, it has been assumed that the weathered rock has sufficient compressive strength to sustain the load of the facility roof and the fill material.

4.1.1.2 Balanced Cut-and-Cover Concept

The balanced cut-and-cover concept (Figure 4-4) is believed to achieve the most economic tradeoff of cost of cover versus cost of fill. The tradeoff between depth of cut to height of cover also permits siting at locations where geological or ground-water problems occur at moderate depths below the surface. Furthermore, if land area constraints are imposed on the plant, then an additional depth of burial adjustment could reduce the areal extent of ground requirements compared to a fully-buried or an aboveground berm configuration. An optimum depth for the reactor containment building could be defined so that ancillary support buildings could be placed adjacent to the cut area on the original ground surface and buried along with the reactor containment building. This arrangement could allow concurrent construction of the ancillary buildings with the reactor containment building and might substantially alleviate schedule penalties accruing to the full underground configuration.

4.1.1.3 Cover-Only Concept

The configuration shown in Figure 4-5 is also categorized as a berm design in that the confining earth material is mounded over the top and around the sides, as shown in the sketch. The construction of the facilities for this design could be very similar to that utilized for a conventional surface plant. The site preparation activities would be almost exactly the same: foundations would be dug and constructed in a conventional manner; the facility buildings, interconnecting water/electrical and control lines, as well as access tunnels would be constructed prior to any backfilling operation. At the completion of the facility construction activities, a specified backfill material would be compacted around and over the structures.

Figure 4-4. Balanced Cut-and-Cover Concept

Figure 4-5. Cover-Only Concept

As in the cut-and-cover configuration, a clay material could be used as an outer impermeable material around the sides, top, and bottom of the berm. The thickness of the confining clay layer probably need not be more than 20 ft on the sides and at the base of the berm, with a 50 ft depth of cover over the facilities. The thickness of the upper layer of confinement material (above the containment structure) is dictated primarily by the low permeability of clay, sufficient thickness being provided to satisfy fission product retention requirements. This layer may not be required, however, if native fill material has adequate permeability and retention characteristics.

If structural considerations are favorable, the location and grouping of the facility structures can be the same as for a surface plant. The major difference dictated by the berm configuration for the cover-only variation is that the buildings must be substantially stronger in order to withstand the confining pressure of the surrounding earthen overburden material. To achieve conditions favorable for facility foundation excavation and construction for the cover-only configuration, the upper 50 ft layer should be of consolidated, non-expansive soils, and the relative density of the material over the complete area should be in excess of 70 percent. For these purposes, the water table should also be below a depth of 50 ft. Suitable borrow areas for sands, gravels, and impermeable clays should be close to the site (preferably within 1.5 mi.). The preferred character of the borrow material is essentially the same as that associated with small earth-filled dams. The preferred local topography would be relatively flat over the construction area (± 5 percent grades). The sides of the excavation have been shown sloped at a ratio of 2:1 for stability.

The design as shown in Figure 4-5 would require $10.8 \times 10^6 \text{ yd}^3$ of borrow material. Approximately $1 \times 10^6 \text{ yd}^3$ would be used for the clay barrier elements while $7 \times 10^6 \text{ yd}^3$ consist of various grades of gravel, sand and silt.

The cover-only berm configuration could represent a reasonable alternative to cut-and-cover configurations in situations where the native

soil material, at depth required for undergrounding, had unsatisfactory geological/strength characteristics or where there was a high water table.

4.1.2 Rock Cavity Configurations

Rock has served as a natural structural material since early man. Both naturally formed and artificially constructed cavities in rock have been used for storage space and occupancy.

As described in a previous section, a large number of hydroelectric stations have been constructed around the world, using the excavated native rock for structures in which to house the power generation equipment. Two basic methods are commonly used for entry into the rock cavity, both of which are also used in underground hydroplant design. Entry to the cavity could be either through a vertical shaft to the cavity or through a horizontal tunnel, depending upon site terrain characteristics. These basic concepts are illustrated in the sketches of Figure 4-6. Variations on either of these concepts are possible depending upon local site and plant functional requirements.

4.1.2.1 Hillside Entry

The concept of a hillside entry includes a horizontal tunnel serving as access to the equipment cavities. Thus, all machinery, equipment, etc., must be brought into the cavity through one or more tunnels of sufficient diameter to accommodate the largest equipment. The diameter of the tunnel is expected to be in the range of 40 ft. The length of the tunnel would be determined by the point of entrance into the cavity, the cavity height, its depth of burial and the slope of the hillside. For designs which call for a specific level of containment, the minimum length of the tunnel(s) would depend upon the slope of the hillside. The sketch of the concept in Figure 4-7 shows results for a 30 percent hillside slope. The cost of the tunnel is a function of its length (for fixed diameter) and the rock characteristics. Therefore, the cost of access tunnels will be a function of the hillside slope, as shown in Figure 4-8. For a cavity buried in a 30 percent slope with a 200 ft depth-of-burial, the cost of excavation of the access tunnel

HORIZONTAL ENTRY

VERTICAL ENTRY

Figure 4-6. Mined Cavity Concepts

Figure 4-7. Containment Tunnel Design - Hillside Concept

Figure 4-8. Tunnel Excavation Cost as a Function of Hillside Slope (for 200 ft Depth of Burial)

would be about \$2 million for excavation costs of \$16/yd³. This cost would be halved for a 34 percent slope. Practical considerations of terrain availability in California for nuclear plant siting suggest that a reasonable range of slope steepness would be between 20 and 45 percent.

4.1.2.2 Vertical Entry

The vertical entry concept might be employed for relatively level surface sites with rock at some depth below surface as shown in Figure 4-9. Shaft entry, including steeply inclined shafts as used in mining applications, must utilize a hoisting mechanism to raise and lower men and equipment between the surface and the plant. Commercial hydraulic lifts of 40,000 lb and traction systems of up to 16,000 lb are available. The high (500 ft) lift heights essentially rule out hydraulic lift systems and traction systems are not large enough to carry all the heavy loads associated with equipment for the NSSS. It appears that new lift concepts must be developed in order to provide a viable vertical shaft entry concept. The most viable access solution is probably a long, spiraled low angle tunnel to reduce site requirements, as shown in Figure 4-9.

Construction of an underground cavity in rock is dependent upon the rock structural properties. The performance of the cavity to contain the fission products and pressurized gases is also dependent in part upon the rock structural properties, especially bulk permeability. Designs for surface PWR reactor containment systems generally include a membrane in the form of a steel liner inside the structural part of the containment to achieve a practically leak-tight container and facilitate cleanup of minor radiation releases. It appears that it would be desirable to retain the steel liner feature for underground mined-rock cavity sited plants. Such a liner would be erected after construction of the cavity and its arch support and might be back-filled to the wall and ceiling as suggested in Figure 4-10. With the use of a steel liner, the natural properties of the rock (especially its permeability and porosity) are no longer the controlling factors for radioactive releases to the atmosphere, unless liner leakage or failure occurs.

Figure 4-9. Containment Tunnel Design - Vertical Concept

Figure 4-10. Possible Anchoring Design for Reactor Cavity Wall

If it is necessary at some point in the accident sequence to utilize the rock surrounding the containment as a medium for fission product entrapment and heat removal, a valving arrangement could be installed in the steel liner to allow controlled passage of the gases and vapors within the containment into the rock. Such valves could be passive devices that open automatically at certain pressure/temperature combinations. It will also be necessary to provide a mechanism to distribute the vented gases after release, outside the steel containment barrier, in order to assure contact of the vented steam and gases with the rock mass over as large a surface area as possible, to maximize heat transfer. Several designs have been considered, including labyrinthian duct work constructed beyond the steel barrier. Another approach would be the use of a porous concrete, similar to the containment concepts for the proposed Ravenwood and Malibu nuclear plants. In both of these proposed surface plant designs, porous concrete was to be placed between inner and outer shells of dual containment barriers. For an underground cavity design concept of this type, the porous concrete material could be placed between the steel liner and the cavity rock. Utilization of supplemental cavities for expansion volumes for the hot gases and vapors could provide another mechanism for energy and pressure management in the event of a Class-9 accident.

4.1.3 Geological Design Background

The ideal rock material for siting and constructing any underground structure would be an elastic, isotropic and homogeneous mass which could be analyzed with theories of continuum mechanics subject to elastic theory. Unfortunately, such idealized materials do not occur in nature. Rock masses, especially the igneous family, are not homogenous, isotropic or elastic. Depending on their origin and geologic history, they usually have been subjected to cooling and solidification (with resulting shrinkage cracks in the case of igneous rocks) and/or subjected to tectonic activity that has fractured the mass for either igneous or sedimentary rocks. In addition, metamorphic and sedimentary rock formations have seldom been produced by sufficiently uniform erosion and depositional processes to be free from

bedding irregularities. As a result, the non-uniform composition of the rock formation must be recognized in the engineering design process and dealt with by appropriate measures. These measures can be economically severe if the siting media is highly fractured so that its bulk strength related properties are badly reduced and it retains few, if any, of the elastic characteristics of the intact, parent rock. In a complex the size of interest to this study, the lining and support requirements under such conditions could be prohibitively expensive.

Terzaghi¹ developed an empirical approach to the lining and support design problem which is illustrated in Figure 4-11. A portion of the overburden material directly over the underground opening (element cdgh) must be carried by roof sidewall supports. The amount of material subjected to support requirements varies depending upon the generalized structural characteristics of the rock formation.

Within the excavation of the tunnel or cavern, the native in situ stresses of the equilibrium rock mass are altered as indicated by Figure 4-11. The rock will fail or yield when the wall stresses (tangential stresses) exceed the unconfined compressive strength of the material. Deere² has developed a relationship for roof and sidewall lining and support requirements as a function of generalized rock mass characteristics as shown in Table 4-1. As indicated, the concept is based upon work originally performed by Terzaghi. In Deere's relationship the structural qualities of the rock mass are defined in terms of composite characteristics defined as a Rock Quality Designation (RQD), which is a field determined index related to fracture frequency and rock mass cohesiveness. Accordingly, conventional unlined structures can be

¹Terzaghi, K., "Rock Tunneling with Steel Supports," by R. V. Proctor and T. L. White, The Commercial Shearing and Stamping Company, Youngstown, Ohio, 1946.

²Deere, et al, "Design of Tunnel Liners and Support Systems," University of Illinois, February 1969.

Figure 4-11. Terzaghi Rock Load Sketch

Table 4-1. Rock Loads and Classifications

FRACTURE SPACING ft.-in.	ROD		ROCK LOAD, H_p		REMARKS
			INITIAL	FINAL	
2 ft. 1 ft. 6 in. 4 in. 2 in. 1 in.	98 95 90 75 50 25 10 2	1 HARD AND INTACT	0	0	LINING ONLY IF SPALLING OR POPPING
		2 HARD STRATIFIED OR SCHISTOSE	0	0.25 B	SPALLING COMMON
		3 MASSIVE, MODERATELY JOINTED	0	0.5 B	SIDE PRESSURE IF STRATA INCLINED, SOME SPALLING
		4 MODERATELY BLOCK AND SEAMY	0	0.25 B TO 0.35 C	GENERALLY NO SIDE PRESSURE ERRATIC LOAD CHANGES FROM POINT TO POINT
		5 VERY BLOCKY, SEAMY AND SHATTERED	0 TO 0.6 C	0.35 C TO 1.1 C	LITTLE OR NO SIDE PRESSURE
		6 COMPLETELY CRUSHED		1.1 C	CONSIDERABLE SIDE PRESSURE. IF SEEPAGE, CONTINUOUS SUPPORT
		7 GRAVEL AND SAND	0.54 C TO 1.2 C	0.62 C TO 1.38 C	DENSE
			0.94 C TO 1.2 C	1.08 C TO 1.38 C	SIDE PRESSURE $p_h = 0.3p(0.5H_t + H_p)$ LOOSE
		8 SQUEEZING, MODERATE DEPTH		1.1 C TO 2.1 C	HEAVY SIDE PRESSURE. CONTINUOUS SUPPORT REQUIRED
		9 SQUEEZING, GREAT DEPTH		2.1 C TO 4.5 C	
WEAK AND COHERENT		10 SWELLING		UP TO 250 ft (80 m)	USE CIRCULAR SUPPORT, IN EXTREME CASES: YIELDING SUPPORT
		TERZAGHI (1946) B IS TUNNEL WIDTH, $C = B + H_t$ - WIDTH + HEIGHT OF TUNNEL FOR CIRCULAR TUNNEL $H_t = 0$			

confidently constructed in rock masses displaying RQDs of 95-100. Lining requirements are minimal with RQDs above 75. For these RQD values, sidewall supports would not generally be required and roof supports could probably be handled with a nominal rock bolting program.

4.1.3.1 Cavity Spans

Preliminary analyses of NSSS equipment dimensions indicate that the required reactor cavity span may be in excess of 140 ft¹, which would be wider than any existing man-made underground structure. However, as indicated in Section 2.3, underground hydroelectric plant chambers have been constructed with spans in excess of 100 ft. Thus, while the span requirements for this study may be larger than those associated with any existing underground structures, any necessary increases will not require substantial advances in the state-of-the-art for achievement. Indeed, there are natural spans that exceed the 140 ft dimension, Rainbow Natural Bridge's span in Southern Utah is 270 ft (sandstone); Mammoth Dome in Kentucky's Mammoth Cave has a span of 150 ft (limestone); and, more significantly, the Carlsbad Cavern maximum span is 625 ft (limestone). However, it should be noted that these natural spans are in sedimentary rock masses, probably due to the increased solubility and erodability of sedimentary materials when compared to igneous materials. Sedimentary rock masses probably approach the ideal of elastic, isotropic, homogenous materials much more closely than most igneous or metamorphic rocks. These natural structures, as well as past experience, underscore the necessity of siting in massive, relatively intact rock formations that represent the upper bounds of rock quality that can be reasonably found and developed.

Recent large-scale siting investigations² have illustrated that a reasonable number of sites are available in California that contain rock masses having the desired RQD and relative size characteristics outlined above.

¹ This dimension has not yet been established with certainty.

² Siting Concepts and Investigations, Ken O'Brien and Associates, USAF, Space and Missile Systems Organization, Norton AFB, California, (Unclassified), April 1969.

4. 1. 3. 2 Critical Physical Properties of Rocks

The material properties of rock formations which are of most significance to the design of underground structures are the Young's modulus of elasticity, compressive strength, density, and seismic-wave velocities. These properties, along with RQD, are the parameters most frequently measured in field investigations. A comparison of field seismic velocity to that obtained from an intact core in the laboratory when related to Young's modulus for the intact sample provides the best information available in early field investigations relative to the fracture frequency and character of joints (degree of tightness) in the rock formation. With values of density, Young's modulus, and compressional (P) or shear (S) wave velocities, all of the other elastic constants customarily used for preliminary design can be estimated by classical methods. Unconfined compressive strengths are normally measured and are utilized to determine support requirements.

4. 1. 3. 3 Igneous and Metamorphic Rocks

Past experience indicates that the granite and diabase families of igneous, intrusive rocks have a high probability of satisfying the design requirements for construction of large underground cavities. Uncertainties with respect to adequate rock properties become greater in the basalt and other flow rock families, based on past investigations of intact rock properties. In general, coarse-grained intrusive material has a higher range of compressive strength and Young's modulus -- two of the most important strength parameters -- than most other rocks. For these materials, compressive strengths can be expected to range from 16,000 to 45,000 psi; while Young's modulus may range from 6 to 17 million psi under adequate site conditions. Basalt, and other intrusive flow rocks, commonly possess compressive strengths ranging from 3,000 to 45,000 psi and Young's moduli of 1×10^6 to 14×10^6 psi.

The data available indicate that the quartzite and gneiss families of metamorphic rocks also have a high probability of yielding satisfactory

sites. The data for competent, intact quartzite indicates ranges of Young's moduli from 1 to 15×10^6 psi and compressive strengths from 17,000 to 56,000 psi. Gneiss ranges from 4.5 to 12×10^6 psi for Young's modulus and 12,000 to 32,000 psi for uniaxial compressive strengths. Flatly foliated schists (45-90 deg) demonstrate a wider range, with Young's moduli from 1 to 6×10^6 psi, and 3,500 to 28,000 psi uniaxial compressive strengths. The steeply foliated schists tend to be somewhat higher in Young's modulus (1 to 12×10^6 psi) but somewhat lower in compressive strengths (2,000 to 20,000 psi). However, steep angles of schistosity offer problems in underground construction that are best avoided if a choice of several sites exists. Data for other metamorphic rock types are available, such as tuff and marble, but they either offer generally unsuitable properties or are not commonly available in large, discrete rock masses in California.

The structural characteristics of metamorphic rocks are quite similar to those of the igneous group. The major difference between the two groups lies in their bulk in situ characteristics. Though metamorphics may sometimes have relatively low strength characteristics, as measured in the laboratory, these weaknesses can sometimes be offset by wider occurrences of massive, relatively intact rock bodies, as long as these large bodies of rock have not been fractured due to intense folding, thrusting, etc.

4.1.3.4 Rock Mass Characteristics

The preferred field values of rock properties are:

Density $\geq 165 \text{ lb/ft}^3$

Seismic Velocity (Compressional
Wave) $\geq 15,000 \text{ ft/sec}$

Young's Modulus $\geq 4 \times 10^6 \text{ psi}$

Unconfined Compressive
Strength $\geq 15,000 \text{ psi}$

A Rock Quality Description (RQD) of ~75 or greater is also desirable. These characteristics correspond to those of massive, moderately blocky

rock bodies. The total bed thickness of the discrete rock body should be approximately 500 ft (or greater) so that the depth-of-cover above the reactor chamber can be at least 200 ft.

From a hydrology standpoint, it is preferable that the site be saturated - but not in an aquifer. This factor is not critical since, if necessary, the site can be artificially watered to produce desired pore water pressures.

Probably the most important factor affecting the suitability of an igneous rock mass for undergrounding is the nature and frequency of its discontinuities; the lower the fracture frequency, (or, conversely, the larger the intact blocks) the better. Cementation may have occurred along fractures, which could "heal" the rock mass, decreasing apparent equivalent fracture frequencies. For ideal conditions, infill material in the cracks under these conditions should be non-soluble in water. The RQD index of ~75 has been empirically related to a mean fracture frequency of about two per foot, allowing for statistical uncertainty in frequency distribution for a given rock mass. There is an inverse relationship between joint spacing and bulk measurements of in situ values of compressive strength and Young's modulus; but this relationship cannot be absolutely quantified until a specific site is selected for a planned facility.

Relaxation of either the strength properties or fracture frequencies of rocks could be permitted for high in situ horizontal stress. Theoretically the horizontal stress would be about six-tenths (0.6) of the lithostatic pressure. This value controls the arching effect over the cavity in a blocky rock mass and thus establishes the requisite nominal depth of rock cover for minimal structural reinforcement of openings. Thus, if the tectonic environment is such that there are significant native in situ horizontal compressive stresses, it may be possible to reduce the depth-of-cover of sound rock for the facility.

Within the igneous and metamorphic formations, highly fractured, blocky rock masses are almost certainly predominant. As indicated, the important factors of these rocks are the frequency and character of fractures

and the orientation of their patterns. On the other hand, sedimentary rocks offer a much higher probability of being free from major vertical discontinuities and imperfections over the probable dimensions of the power plant caverns. Here, experience indicates that the major problems could well be variations of in-bed thickness and consequent local variations in structural properties. However, in some sedimentary rock formations, these imperfections may be more easily dealt with than vertical fractures in igneous or metamorphic rocks. The measured laboratory strength properties of intact sedimentary rock specimens are generally much lower (on the average) than igneous and metamorphic sample measurements. However, the overall bulk in situ material properties for sedimentary formations may run higher than those of an equivalent igneous or metamorphic site due to the general homogeneity of the total mass of the sedimentary bed.

Three major classes of sedimentary rocks appear to be of greatest interest to the siting effort: sandstones; limestones; and dolomites. Although limestone and other carbonate bearing rocks may be rejected because of CO_2 generation when subject to very high temperatures, from a strength standpoint, the limestones and dolomites appear to have very attractive characteristics. Their compressive strength envelopes range from 7,000 to 35,000 psi, with corresponding Young's moduli ranging from 3.5 to 13×10^6 psi for intact specimens. Sandstones, as might be expected, have a much greater range of compressive strength (i. e., 1,000 to 45,000 psi) but generally lower values for Young's moduli (i. e., 0.4 to 7×10^6 psi).

When dealing with spans of the order of 140 ft for the reactor chamber, the expected increased cohesiveness and homogeneity of a sedimentary rock mass could conceivably offset the higher strength values demonstrated in the igneous and metamorphic rock cores.

A total bed thickness of 500 ft is still appropriate for structures in sedimentary rocks. Two or more types of formations in adjacent beds could be used to provide the required site thickness, providing that each has the desired strength qualities and that the bedding interfaces are thick and have smooth (rather than abrupt) transitions.

4.2 DESIGN FEATURES AND SYSTEMS DESCRIPTIONS OF SURFACE NUCLEAR POWER PLANTS

The harnessing of nuclear energy in the early 1940s led to the development of numerous power reactor concepts. Among these, two forms of the light-water reactor (LWR), the pressurized water reactor (PWR), and the boiling water reactor (BWR), showed technological promise and have achieved commercial status. Numerous PWR and BWR plant design variations have evolved. Other reactor concepts, such as the high-temperature gas reactor (HTGR), have appeared on the commercial horizon; but none has achieved substantial market penetration.

In January 1974, there were 207,626 MW_e of domestic U.S. nuclear generation capacity which were either operational, under construction, or had been committed for construction. At that time, the BWR represented 34 % (70,019 MW_e), the PWR 62 % (129,567 MW_e), and the HTGR slightly under 5 % (8,040 MW_e) of the total nuclear commitment. Since 1974, substantial reductions in committed nuclear capacity have occurred. At present, only one 330 MW_e HTGR facility has been constructed. No others have been committed for development. As of this writing, only LWRs are commercially available in the United States. It is possible that this situation may change soon, however, pending the findings of an HTGR commercialization study presently underway. Consequently, The Aerospace Corporation study was directed at reactor systems which are currently marketed in the United States, the PWR and the BWR. That other reactor types have not been considered does not imply that value judgements have been made with respect to their relative safety or technical feasibility.

Given the constraints of this study, it was judged prudent to emphasize one LWR design, the PWR, which was felt to be representative of current, as well as practical, future designs, to serve as a baseline. Thus, primary attention has been given to the evaluation of this baseline reactor with sensitivity studies performed to analyze the features of the BWR which affect its capacity as a nuclear steam supply in an underground facility.

4.2.1 Reactors in California

There are currently three operating power reactors in California, Humboldt Bay,¹ 70 MW_e, PGE; San Onofre 1, 430 MW_e, SCE/SDGE; Rancho Seco 1, 916 MW_e, SMUD. In addition to these, there are four plants either under construction: Diablo Canyon 1 and 2, 2120 MW_e, PGE; San Onofre 2 and 3, 2280 MW_e, SCE/SDGE; or in various stages of review, including Sundesert 1 and 2, 1860 MW_e, SDGE; San Joaquin 1 through 4, 5200 MW_e, LADWP. The current operational nuclear capacity is 1,416 MW_e, the capacity under construction is 4,400 MW_e, and the capacity proposed is 7,060 MW_e for a grand total of 12,876 MW_e. All but 70 MW_e (Humboldt Bay) of the operational plants and those under construction are PWRs. The proposed 930 MW_e capacity units of the Sundesert plant have been designated PWRs, but the type of reactor for the 1300 MW_e units of the San Joaquin Project is still undecided. At present, the PWR almost totally dominates the California nuclear power plant picture. However, the BWR was considered to be a reasonable candidate for construction, and therefore, was included for review in this study, though the PWR was chosen as the baseline reactor for investigation.

4.2.2 Standard Reactor Plant Designs

Early in the 1970s, the U.S. Atomic Energy Commission (AEC) indicated a desire to have the reactor vendors standardize the nuclear steam supply system (NSSS) design so that a common safety analysis report would be applicable to each NSSS built to the specifications of that standard design.

The several vendors of U.S. LWR systems have developed standard reference reactor designs. The manufacturers of these reactors and their design designations are shown in Table 4-2. Only reactor designs with a capacity of at least 1,000 MW_e have been shown in the table. The NSSS safety analysis report by Westinghouse, called "RESAR", was the first of such analyses submitted in response to AEC direction. Work has also been initiated by several major architect/engineering firms with the support of the U.S. Nuclear Regulatory Commission (NRC), the agency which

¹The Humboldt Bay reactor is currently being considered for relicensing.

Table 4-2. Reference Light-Water Reactor Designs

Design	Power Level		NSSS Vendor
	Net Electric (MW _e)	Reactor Core (MW _t)	
RESAR - 41	1295	3800	Westinghouse Electric
System - 80 (CESSAR)	~1295	3800	Combustion Engineering
B-SAR-241	1322	3800	Babcock & Wilcox
BWR/6, 238-732 (GESSAR)	1220	3579	General Electric

took over the regulatory responsibilities of the AEC, to develop reference designs for the remaining portions of the plant. The NRC's philosophy behind the standardized balance-of-plant (BOP) designs was to speed up the application procedures for the review and licensing process for nuclear plants, as well as to allow more attention to be directed to safety matters by standardizing as many elements of the plants as possible. This would tend to reduce the requirement for repeated review of nearly identical features of similar facilities.

The first "reference plant" design using a PWR was submitted to the U.S. Atomic Energy Commission for approval in 1974. This was the Stone and Webster (S & W) balance-of-plant design (Ref. 4-1) based on the Westinghouse Electric NSSS (RESAR-41) (Ref. 4-2). This design has also been adapted to the Babcock and Wilcox B-241 and Combustion Engineering System-80 nuclear steam supply systems. These combined NSSS and BOP designs reportedly incorporate the latest concepts in PWR plant and safety systems. Other reference PWR-BOP designs which have been submitted to the NRC for regulatory approval include:

Braun Standard Turbine Island

Fluor Pioneer BOP - BOPSSAR

The material presently available on many of these designs is only partially complete. Therefore, because the S & W design was complete, readily available, and approved by NRC, it was selected for analysis as the base-line surface plant for the Aerospace preliminary studies, due to the brief period available for this review. It should be noted that the reference designs for the S & W BOP would have to be modified to meet California seismic criteria.

A reference BWR nuclear plant design is also available (marketed only by General Electric), termed the Integrated Nuclear Island (GESSAR-238, 251) (Ref. 4-3). It includes steam generation, fuel handling, and waste disposal facilities. The standardized G. E. design uses a BWR-6 NSSS containment (Ref. 4-4). The G. E. reference plant concept, currently in the

process of review, is considered to offer a more compact and efficient arrangement than earlier BWR balance-of-plant designs. Therefore, this concept was selected as the alternative LWR for the Aerospace evaluation of underground plant configurations for BWR reactors.

A diagram from the S & W reference design showing the general plant layout is given in Figure 4-12. A corresponding diagram of plant layout and equipment arrangement is shown in Figure 4-13 for the G. E. Integrated Nuclear Island concept. From the description and engineering specifications provided, dimensions for each of the major plant structures were derived. These are tabulated for both BWR and PWR systems in Table 4-3.

The overall dimensions for the BWR containment and auxiliary buildings are shown in the perspective sketch of Figure 4-14, while similar information for the S & W PWR nuclear system is shown in Figure 4-15. In both cases, only the structures considered most likely to yield benefits from undergrounding are shown. The basis for selecting those structures considered to be the most promising candidates for undergrounding is discussed in Section 4.4.

4.2.3 Definition of Plant Systems and Interfaces

Information on the particular technical function of each subsystem of a nuclear plant was compiled for examination. As indicated in Tables 4-4 and 4-5, a total of 30 such subsystems were identified for the PWR plant and 26 for the BWR/6. Neither the electrical power generation equipment nor its subsystems is shown for either the PWR or the BWR systems. It was decided that there is sufficient commonality of equipment and interfaces for PWR and BWR turbine generators that a separate investigation for each was not warranted for the purpose of this study. The complete inventory and descriptions of all NSSS related power plant systems and piping interconnections are given in Appendix II. A summary of the interconnection information for PWR and BWR plants (Reference Plant Piping Inventory) is shown in Tables 4-6 and 4-7, respectively.

Plan View

Elevation View

Figure 4-12. Stone and Webster Standard PWR Balance-of-Plant Design

REACTOR BUILDING

1. SHIELD BUILDING
2. FREE-STANDING STEEL CONTAINMENT
3. POLAR CRANE
4. REFUELING PLATFORM
5. UPPER POOL
6. REACTOR WATER CLEANUP
7. REACTOR VESSEL
8. STEAM LINE
9. SHIELD WALL
10. FEEDWATER LINE
11. DRYWELL
12. RECIRCULATION LOOP
13. WEIR WALL
14. HORIZONTAL VENT
15. SUPPRESSION POOL

AUXILIARY BUILDING

16. STEAM LINE TUNNEL
17. MOTOR CONTROL CENTERS
18. RHR SYSTEM

FUEL BUILDING

19. FUEL TRANSFER BRIDGE
20. FUEL TRANSFER TUBE
21. CASK HANDLING CRANE
22. FUEL STORAGE POOL
23. NEW FUEL VAULT
24. CASK LOADING POOL
25. SPENT FUEL SHIPPING CASK
26. FUEL CASK SKID

Figure 4-13. Reference BWR/6 Mark III Containment

Table 4-3. Underground Plant, Basic Systems Envelope Physical Dimensions

Plants and Elements	Geometric Shape	Dimensions		
		height(ft)	width(ft)	length(ft)
<u>PWR PLANTS</u>				
<u>Possible Underground Elements</u>				
Reactor Containment (Unmodified)	vertical right cylinder with elliptical dome	275	140 foot inside diameter	
Reactor Containment (Modified)	rectangular	177	96	108
Fuel Handling & Storage	rectangular	80	65	175
Liquid & Gaseous Waste	rectangular	86	75	600
Solid Waste	rectangular	62	40	118
Turbine Generator	rectangular	122	170	300
<u>Surface Elements</u>				
Control Room	rectangular	75	115	115
Emergency Room (Diesel Generators)	rectangular	30	115	90
Service & Maintenance	rectangular	56	85	230
<u>BWR PLANTS</u>				
<u>Possible Underground Elements</u>				
Reactor Containment (Unmodified)	vertical right cylinder with elliptical dome	210	130 foot inside diameter	
Reactor Containment (Modified)	rectangular	237	74	110
Fuel Handling & Storage	rectangular	120	45	170
Auxiliary Equipment	rectangular	140	85	170
Turbine Generator	rectangular	73	200	400
Liquid and Solid Waste	rectangular	85	105	150
<u>Surface Elements</u>				
Control Room	rectangular	55	60	70
Emergency Power (Diesel Generators)	rectangular	30	75	98
Service & Maintenance	rectangular	50	150	185

Figure 4-14. Reference BWR Nuclear Island Containment Structure

Figure 4-15. Reference PWR Nuclear Island

Table 4-4. PWR Reference Plant Systems

Aerated Vent and Drain
Annulus Building HVAC
Auxiliary Feedwater System
Auxiliary Steam and Condensate
Boron Recovery
Carbon Dioxide Fire Protection
Containment Atmosphere Air-
borne Radiation Monitoring
Containment Purge Air
Condensate
Condenser Evacuation
Control Building HVAC
Cable Spreading Room
Ventilation
Control Room Air
Conditioning
Refrigeration Equipment
Room Air Conditioning
Diesel Generator Room
Ventilation
Electrical Tunnel Ventilation
Emergency Switchgear Air
Conditioning
Demineralized Water Makeup
Feedwater
Fuel Storage (New and Spent)
Fuel Building HVAC
Fuel Pool Purification
Gaseous Vent and Drain
Instrument and Service Air
Main Steam
Primary Grade Water
Radioactive Liquid Waste
Radioactive Solid Waste
Reactor Plant Component
Cooling Water
Reactor Plant Gas Sampling
Reactor Plant Gas Supply
Reactor Plant Service Water
Reactor Plant Ventilation
Radioactive Solid Waste and
Decontamination HVAC
Steam Generator Blowdown
Turbine Building HVAC
Water Fire Protection
Chilled Water

Table 4-5. BWR/6 Reference Plant Systems

Auxiliary Building Electrical
Areas, Corridors, Steam
Tunnel and Elevator Tower
HVAC
Auxiliary Building Emergency
Core Cooling System Area
Pressure Control
Carbon Dioxide Fire Protection
Clean Radwaste Drain
Closed Cooling Water
Control Building Chilled Water
Control Building HVAC
Control Building Potable Water
Demineralized Water and
Condensate Distribution
Demineralized Water and
Condensate Distribution
Detergent Drain
Diesel Generator and
Switchgear Rooms HVAC
Diesel Fuel Transfer
Dirty Radwaste Drain
Essential Service Water
Fuel Building HVAC
Fuel Pool Cooling & Cleanup
Heated Water Distribution
High Pressure Core Spray
Service Water
Hydrogen Mixing, Drywell
Relief and Containment
Vacuum Relief
Pneumatic Supply
Radwaste Building HVAC
Reactor Island Non-Essential
Chilled Water
Service Air Distribution
Standby Gas Treatment (SGTS)
Wet Standpipe Fire Protection

Table 4-6. PWR Reference Plant Piping Inventory

Building or Area Interconnected

Number of Lines

Annulus Building with -

Atmosphere	4
Balance of Plant (BOP) water	1
Containment Structure	58*
Control Building	3
Diesel Generator Building	2
Fuel Building	6**
Radioactive Solid Waste and Decontamination Building	48***
Turbine Building	19
Ultimate Heat Sink	15
Yard	8

Containment Structure with -

Atmosphere	1 (air exhaust)
Turbine Building	2

Footnotes

- * Includes four lines for Main Steam Line interface, four lines for Feedwater System interfaces, and four lines for Steam Generator Blowdown System interfaces shown in Table 7 (Ref. 1); also includes 2 air exhaust ducts.
- ** Includes two air intake and two air exhaust ducts.
- *** Includes one air exhaust duct.

Table 4-6. PWR Reference Plant Piping Inventory (Continued)

<u>Building or Area Interconnected</u>	<u>Number of Lines</u>
Control Building with -	
Atmosphere	18 (9 ea. air intake and exhaust ducts)
Balance of Plant (BOP) water	1
Normal Switchgear Building	1
Diesel Generator Building with -	
Balance of Plant (BOP) water	1
Fuel Building with -	
Balance of Plant (BOP) water	1
Radioactive Solid Waste and Decontamination Building	5
Yard	1
Radioactive Gaseous Waste Building with -	
Turbine Building	2
Radioactive Solid Waste and Decontamination Building with -	
Atmosphere	1 (air intake duct)
Balance of Plant (BOP) water	1
Yard	2
Turbine Building with -	
Atmosphere	18 (4 air intake and 14 air exhaust ducts)
Auxiliary Boiler Building	3
Balance of Plant (BOP) water	1

Table 4-6. PWR Reference Plant Piping Inventory (Continued)

<u>Building or Area Interconnected</u>	<u>Number of Lines</u>
Turbine Building with - (continued)	
Service Building	4
Yard	4

Table 4-7. BWR/6 Reference Plant Piping Inventory

<u>Building or Area Interconnected</u>	<u>Number of Lines</u>	<u>Size (in)</u>
Auxiliary Building with -		
Atmosphere	7 (3 air intake ducts and 4 air exhaust ducts)	
Balance of Plant (BOP)	2	3
	4	4
	2	6
	2	10
	1	18
Containment Structure	2	24
	4	1
	4	1-1/2
	9	3
	5	4
	3	6
	1	8
	1	18
	1 air exhaust duct	-
Control Building	1	1
	2	2
	4	3
	4	6
Diesel Generator Buildings	2	1
	1	2

Table 4-7. BWR/6 Reference Plant Piping Inventory (Continued)

<u>Building or Area Interconnected</u>	<u>Number of Lines</u>	<u>Size (in)</u>
Auxiliary Building with - (continued)		
Fuel Building	5	3/4
	5	1
	2	1-1/2
	2	2
	8	2-1/2
	1	3
	5	4
	2	6
	5	10
	2	14
	1 personnel airlock	-
	1 air exhaust duct	-
Radwaste Building	2	6
	13	3
Turbine Building	1	4
Containment Structure with -		
Balance of Plant (BOP)	1	12
Fuel Building	8	3/4
	8	1
	5	2
	1	3
	1	4
	1	6
	1 air exhaust ducts	-

Table 4-7. BWR/6 Reference Plant Piping Inventory (Continued)

<u>Building or Area Interconnected</u>	<u>Number of Lines</u>	<u>Size (in)</u>
Control Building with -		
Atmosphere	2	2
	3 (2 air intake ducts and 1 air exhaust duct)	-
Balance of Plant (BOP)	1	3
	2	4
	2	6
Division 1 Diesel Generator Building	1	1-1/2
Diesel Generator Buildings with -		
Atmosphere	2 (1 each air intake duct and 1 air exhaust duct)	-
Balance of Plant (BOP)	1	1-1/2
	1	2
	4	2-1/2
	2	4
	2	8
Drywell to Atmosphere	1 air exhaust duct	-

Table 4-7. BWR/6 Reference Plant Piping Inventory (Continued)

<u>Building or Area Interconnected</u>	<u>Number of Lines</u>	<u>Size (in)</u>
Fuel Building with -		
Atmosphere	2 air exhaust ducts	2
Balance of Plant (BOP)	2	4
	2	6
	4	10
	5	12
	1	24
	2	-
Containment Structure/Shield Building Annulus	1 air intake duct	-
Fuel Building with -		
Diesel Generator Buildings	1	1
	2	1-1/2
	1	2
Radwaste Building	4	3
Radwaste Building with -		
Balance of Plant (BOP)	1	3
	6	6
Atmosphere	5 (2 air intake ducts and 3 air exhaust ducts)	-
Switchgear Building to Atmosphere	2 (1 ea. air intake duct and air exhaust duct)	

The large number of systems and the diversity of their contributions to the safe operation of a nuclear plant heightens the need for close attention to the grouping and layout of the system components and their housing within the structures of the facilities. It was observed that while some systems (e.g., radwaste, and spent fuel storage) could conceivably be placed above ground without increasing the hazards, it appeared to be preferable to place these systems underground to achieve substantial simplification of the plant design and operation. As a design goal, the underground location for each system should be maintained as close as practical to its location in the standardized surface plant design.

The NRC has established seismic design criteria for structures, components, equipment, and systems. These criteria apply to two levels of the earthquake acceleration loadings, the "safe shutdown" and "operating basis" earthquakes. Currently, the majority of the reference plant designs (including the Stone and Webster SWESSAR-P1) specify a 0.3g peak acceleration for the safe shutdown earthquake (SSE) and a 0.15g peak acceleration for the operating basis earthquake (OBE). These values for the SSE and OBE are suitable for most parts of the U.S. except California. Values for California would have to be based upon specific site characteristics. To assure adequacy of designs, the equipment and structures to which these criteria apply are defined as Seismic Category I elements of the facility design. These Category I elements of the plants are presented in Tables 4-8 and 4-9 for PWRs and BWRs. In the course of deciding which structures to place underground, the Category I items were given first consideration in order to obtain the advantage of the lower seismic acceleration levels at depth. If, in the course of the evaluation, however, the complexity or cost of system interconnections were increased by undergrounding, then a trade-off analysis was undertaken. In most instances, a decision was easily made on a judgemental basis. The structures and systems selected for undergrounding are shown in a subsequent section.

Table 4-8. PWR Seismic Category I Structures,
Systems, and Miscellaneous Components

STRUCTURES

Containment
Cable Tunnel from Containment Structure to Control Building
Auxiliary (Annulus) Building
Control Building
Diesel Generator Building
Diesel Generator Fuel Oil Pump House
Reactor Vessel Support Shield Tank

SYSTEMS

Containment Spray
Containment Isolation
Combustible Gas Control
Electrical
Fuel Pool Cooling and Purification
Reactor Plant Service Water
Reactor Plant Component Cooling Water
Reactor Plant Sampling
Air Conditioning, Heating, Cooling, and Ventilation
Diesel Generator Fuel Oil Storage and Transfer
Main Steam
Feedwater
Steam Generator Blowdown
Auxiliary Feedwater
Radioactive Gaseous Waste
Containment Atmosphere Radiation Monitoring

MISCELLANEOUS COMPONENTS

Containment Structure Polar Crane
Fuel Building Crane

Table 4-9. BWR Nuclear Island Seismic Category I
Structures and Systems

STRUCTURES

Nuclear Island

Reactor Building

Shield Building

Containment Vessel

Drywell

Reactor Pedestal

Fuel Building

Auxiliary Building

Diesel Generator Building

Control Building

Radioactive Waste Building Substructure

SYSTEMS

Diesel Generator Fuel Oil Storage Tank Vault

Cooling Water Intake Structure

Condensate Storage Tank Basin

4.3 ANALYSIS OF PLANT DESIGN/COST ESTIMATING ACCURACY

Decisions as to the practicality of subsurface nuclear plants will ultimately be based in part on comparisons of their accident mitigation benefits and implementation costs. Cost-benefit ratios are no more accurate as measures for comparing alternative nuclear plant configurations than the estimates of costs for construction of the facilities. All estimates of costs for future projects are uncertain to some degree. This section of the report analyzes the nature and effects of these uncertainties in order to provide a basis for managing design efforts of architect-engineers in the Detailed Study portions of the program. The goal of these analyses was to develop cost estimates that were as accurate as possible, consistent with the available data and fiscal resources of the program.

The analysis of estimated costs and related uncertainties emphasizes differences between subsurface and surface nuclear plants. A fundamental postulate was taken that to fully benefit from the experience gained from surface plant installations, wherever feasible, modifications to the configurations of subsurface plant subsystems and equipments should be minimal. On this basis, modifications made necessary by subsurface space restrictions, or functional or environmental differences, would be designed for minimum cost impact (including the impact of changes on the costs associated with regulatory approval, installation, and testing).

For purposes of comparison of subsurface and surface installations, cost estimates developed by the AEC for a surface PWR plant of 1000 MW_e capacity (Ref. 4-5) were used as a baseline for surface facilities. The cost impact of installing this baseline system underground was estimated in order to identify the magnitudes of cost differences to be expected between surface and underground facility construction. The uncertainties relating to the cost differences were estimated as a basis for determination of the range of uncertainty of total plant costs.

As reported in this section, the uncertainties in plant cost estimates associated with design of a facility (of the type to be conducted in the Detailed Study portion of the ERCDC program) were found to be dominated

by inexperience factors. On this basis, estimates have been made of design contributions to the overall cost estimation uncertainty. Emphasis was placed upon the contributions of uncertainties related to the design in order to evaluate the relative impact of various elements of the engineering design of the facility. Results of the evaluation of the individual contributions can be used as a basis for determining the emphasis to be placed on reducing the design uncertainties in each of the design accounts. Of course, other factors contribute to the overall uncertainties for total cost estimates of underground facilities. These factors have also been discussed in a qualitative manner in this section, but the emphasis has been placed upon the design contributions, as noted. Estimates of relative total cost impacts of several underground concepts have also been provided (cf Table 4-15). More detailed estimates of uncertainties in overall plant costs with greater emphasis upon non-design related uncertainties will be conducted by other contractors during the Detailed Study portion of the ERCDC program.

The analysis of cost estimates developed by A/Es on the basis of conceptual designs and their uncertainty is discussed below in the following order:

- a. Factors and issues which will affect cost estimates and their accuracy are discussed in a qualitative manner.
- b. Historical records of cost estimating accuracy for nuclear plants are reviewed.
- c. Engineering design contributions to the uncertainty of cost estimates for underground facilities are analyzed.
- d. Plant subsystem design detail and cost accuracy relationships are discussed.

4.3.1 Cost Estimating Issues and Factors

It was initially necessary to distinguish those cost factors and issues which affect estimating accuracy for all nuclear plants (in general)

from those that are specific for subsurface plants. Only the latter are pertinent to decisions to install plants on the surface or below the surface.

Several factors have historically caused wide differences between initial cost estimates and final actual costs. Nuclear plants are complex, and the standards and regulatory procedures necessary for their control are still evolving. Construction experience for nuclear plants is limited, when compared to fossil-fueled plants, and most of this experience is with plants designed and constructed to earlier regulatory standards. Moreover, price levels have been unstable during the periods of construction for the more recent (and pertinent) commercial nuclear plants.

Some uncertainty is unavoidable in estimating a complex project, even where the plant design is replicated and price levels are stable. Uncertainty is introduced by variation in size related factors, labor productivity, management skill, and the impact of contingent events (late deliveries, unusually severe weather effects, and financial problems, to name a few) which cause schedule delays and cost increases due to their effect on labor efficiency. In estimating facility costs, contingency factors are normally applied to allow for the cost uncertainties due to these variations. When the contingency factors are based on extensive experience with very similar projects, their application may be adequate. It should be noted, however, that conditions usually assumed for initial architect-engineer estimates of overall plant costs customarily exclude the effects of factors such as strikes, failure of contracting agencies to provide timely services, or delays in authorization to proceed where such events are beyond the contractor's control. Such cost effects are normally covered by contractual provisions. Contingency factors do not cover events of this type which are deliberately excluded from the cost estimates.

Uncertainty is increased when the system being estimated includes new or modified designs or construction standards. This is the condition prevailing with essentially all nuclear power plant construction to date. New standards must be translated into system and equipment designs

and/or new installation procedures and schedules. Cost effects which are perceived can be estimated; however, with new designs and procedures, errors and omissions must be expected. Their correction increases costs directly, may have significant impacts owing to redesign and rework, and may cause schedule disruption of other construction activities. This tendency for initial production models to be high in cost in comparison with subsequent units is the commonly recognized "experience curve" effect.

Finally, attempts to forecast future prices and interest rates in order to estimate ultimate costs, including interest charges, in current dollars, are uncertain where projects extend a decade (or more) into the future.

The areas of cost estimation uncertainty related specifically to subsurface installations were emphasized in this study. These areas of difference are those related to (a) differences in excavation and construction, (b) equipment design modifications, (c) installation of equipment in restricted quarters, (d) total project time, and (e) extrapolation of data from experience. These areas and their general impacts on uncertainty in cost estimates are discussed below.

4.3.1.1 Excavation and Construction

For plants installed in excavated rock cavities, excavation costs will be functions of the volumes and the geometry of volumes required for subsurface chambers, tunnels, and access shafts or adits; depth of burial; amount of overburden; the type of rock and its structural properties; and the amount of subsurface water encountered.

Similarly, excavation costs for berm-covered installations will be functions of the (a) plant configuration and depth of burial, (b) site geography and geology, (c) excavated material characteristics, (d) availability of suitable fill material, and (e) the amount and depth of subsurface water encountered.

The choice of an actual location is influenced by factors such as local topography, land use, ownership, and transportation facilities, as well as proximity to electrical loads. The availability of sites for which both these factors and excavation factors are favorable is not now known.

Large uncertainties in excavation cost estimates can be expected where they are based only on subsurface characteristics implied from surface evidence. Reducing this uncertainty requires an extensive geological survey of the selected site to be conducted with core drilling and testing performed to determine subsurface conditions. Although geological surveys are required for surface-sited facilities, they will have to be more complete for underground construction, and consequently, more expensive.

Construction costs for underground structures will differ from surface construction with respect to quantities of materials required and with respect to accessibility for construction. Berm-covered buildings are subject to different design loads and design techniques than those associated with surface buildings. In construction of mined-rock facilities, suitably lined, rock cavities replace surface building exterior structures. It has been assumed that configurations of structures within the buildings will be much the same as surface configurations except for differences resulting from different seismic criteria. Material quantities may be determined by preliminary design with relatively small error. The major cost uncertainty is related to accessibility for construction, particularly for construction in rock cavities. It is estimated that limited access and workspace will increase labor costs (e. g., labor costs per pound of reinforcing steel or per cubic yard of concrete in place) by a significant amount. It should be noted that underground construction experience, especially with the berm concept, is not as mature as above ground experience. Hence, from an estimating standpoint, underground work is at an earlier point on the "learning curve".

4.3.1.2 Equipment Design Modifications

Equipment and subsystem configurations were assumed to be unchanged for subsurface installation, except where necessary because of span limitations and cavity or building separation requirements. Therefore, the major changes which affect cost appeared to be the increased lengths of lines interconnecting subsystems, such as for coolant, steam supply, condenser water, electric power, and control system lines. Condensers installed in subsurface facilities might require redesign to withstand higher water heads where the cooling water source is above ground. On the other hand, modified seismic design criteria might also allow redesign of equipment to reduce costs; however, the cost of redesign, test, and approval of new equipment and establishing quantitative estimates of confidence in its reliability could more than offset the reduced fabricating costs associated with seismic benefits.

Environmental differences below ground-temperature, humidity, pressure - could affect some equipment, but the cost effects appear to be slight.

4.3.1.3 Reduced Access and Work Space

Access to, and workspace within, rock cavities is more restricted for underground installations and will increase costs. With surface plants, a number of access areas through walls or roofs of buildings can be left open until equipment installation is complete. With this arrangement, some fabrication and assembly of equipment can be completed outside the buildings. Internal assembly problems can be reduced under these circumstances, and the scheduling of material flow from outside areas to installation areas is apparently simpler and requires less material storage in the installation areas than appears to be the case for underground construction. With subsurface cavities, in contrast, access will be limited to excavated tunnels and shafts. The practicability of external assembly and transport to installed locations will be reduced and work area congestion is expected to be increased by the larger proportion of required subsurface work.

Material flow from outside will take more time, and its scheduling will be less certain, requiring increased storage in the installation areas and causing further crowding of work spaces. The increased complexity of material handling, storage, and scheduling, along with reduced access and work space, and the increased opportunity for material interference between installations in the same area, increase the likelihood of schedule disruption, delay, and reduced efficiency of labor.

If equipment installation can be completed prior to backfilling operations, access to the berm concepts should be similar to surface plants. If this is not possible, then it would apparently suffer the same disadvantages as a rock cavity design.

4.3.1.4 Schedule Duration

Increases in the duration of construction periods affect total costs by increasing interest charges during construction. The effect of increased schedule time is significant. The addition of one year to a schedule, with no change in basic constant-dollar costs, increases total costs by, approximately, the basic cost times the average of escalation and interest rates.

On the basis of experience with underground hydroelectric facilities, schedules for rock-cavity installations were estimated to require about one additional year, owing to the longer time required for excavation. Berm-covered installations will probably require somewhat less additional time.

4.3.1.5 Experience Factor

Each repetition in the performance of a complex task costs progressively less than the first. This "learning" curve effect results from progressive improvement in methods and procedures gained from increased experience. Compared to surface installations, subsurface plant construction requires changes in plant configurations, some new equipment design, and development of new installation procedures. The new plans, detailed installation procedures, and schedules which must be developed, will include a

number of undetected errors and ambiguities. These will disrupt the smooth flow of scheduled activity, resulting in scrapping and rework, a reduction of labor productivity, and schedule slippage.

Many major equipment items for the nuclear power plant will be modified very little for underground installation. Costs for these can be estimated with reasonable accuracy. Similarly, quantities and costs of other relatively unchanged materials can be estimated with confidence.

The major uncertainty is the effect of the new subsurface conditions on the estimated costs (i. e., productivity) of installation labor. Two adjustments to experience factors in labor-cost relationships are involved: the first compensates for decreased labor efficiency due to restricted access and workspace, and the second takes into account first-time installation of underground equipment and material.

The tendency in planning and estimating new projects is to allow for the perceived effects of changes from experience. Effects not initially perceived are commonly underestimated; i. e., contingency factors to allow for the unperceived effects tend to be too small. Thus, actual costs for projects with a high proportion of new procedures tend to exceed initial estimates. This effect is discussed in more detail in Section 4.3.3.

4.3.2 Historical Record of Cost Estimating Accuracy

The historical record of the accuracy of initial estimates of costs for nuclear-electric plants provides a valuable source of data on which to base estimates of the overall uncertainty in initial cost estimates of subsurface plants.

In compiling these data, both the initial estimates and the actual costs in the historical data have been expressed in current (i. e., variable) dollars to include the effects of inflation. Differences between initial estimates and final costs which result only from increases in the rate of inflation have been excluded for purposes of analysis. The accuracy of estimating future inflation rates would affect subsurface and surface plants equally. As adjusted, and stated in constant dollars, the initial estimates

and final costs pertain to basic capital costs,¹ in terms of the same (constant) dollars, excluding the effects of changes in prices and interest rates.

The ratio of these basic capital costs - final costs to initially estimated costs - expresses the accuracy of initial estimates of the basic capital cost.

4.3.2.1 Early Experimental and Prototype Plants

Cost estimates for experimental nuclear plants completed in 1956 and 1957 and prototype plants completed between 1957 and 1963 are shown in tabular form in Figure 4-16. These data are expressed in current dollars per kW_e. They were not adjusted for the effects of inflation, because inflation rates during this period were low and their effect on estimates was small.

The data are grouped in three categories: (a) small (1.0 to 10 MW_e); (b) larger experimental plants; and (c) commercial prototype plants (all larger than 100 MW_e).

Ratios of estimates - final cost to initial estimate - calculated for each plant, vary from 1.09 to 2.70. Averages of these ratios for each of the three groups taken individually show much less variation: 1.80 for the small experimental plants, 1.93 for the larger experimental plants, and 1.42 for the commercial prototypes. Ratios for the larger experimental and prototype plants are quite possibly low. The final costs are the latest determined as of the date of the reference and do not necessarily represent completed project costs. Somewhat higher final costs may have been experienced, judging from the general tendency of estimates of final cost to increase from start to completion of the project.

The general level of uncertainty for these early plants can be represented by a ratio of 1.5 to 2.0.

¹ "Basic capital costs" is used herein in the special sense as defined. In common usage, "capital costs" are usually stated in current (variable) dollars, and interest during construction is included.

Experimental and Prototype Plants

Designation	Size, MW _e	Year Completed	Orig. Est., \$/kW _e	Actual on (1957 Est., \$/kW _e	Cost Ratio, Actual/Orig
ERWR Argonne Ill.	5	1956	710	1000	1.41
SRE Santa Susana, Ca.	6	1957	575	1550	2.70
HRE-2 Oak Ridge	1	1957	1700	3400	2.00
APPR Fort Belvoir Va.	1.75	1957	2200	2400	1.09
					Avg. 1.80
<u>Larger Experimental</u>					
RCPA ^{1/}	19	1962	260	(650)	(2.50)
CPPD ^{2/}	75	1962	325	(570)	(1.75)
Piqua ^{3/}	12	1963	420	(650)	(1.55)
					Avg. (1.93)
<u>Prototype Commercial</u>					
Shippingport (PWR)	160	1957	680	1040	1.53
Yankee (PWR)	125	1960	250	(350)	(1.40)
Indian Point (PWR)	160	1962	245	(325)	(1.33)
					Avg. (1.42)

Source: Nucleonics, January 1958

^{1/} Rural Coop. Power Ass'n, Elk River, Minn. (Boiling Water).

^{2/} Consumers Public Power Ass'n, Hallam, Nebr. (Sodium graphite).

^{3/} City of Piqua, Ohio (Organic cooled and moderated).

Figure 4-16. Early Cost Estimating Experience

4.3.2.2 Full-Scale Commercial Plants

Cost estimates for commercial plants announced in the years 1965 through 1970, from Ref. 4-6, are shown in Figure 4-17. These data exclude 11 plants ordered in 1965, 1966, and 1967 on "turnkey" contracts, for which the true costs are not available.

Original estimates, and mid-1974 estimates, are tabulated in the figure, corresponding estimates of basic capital costs (excluding price escalation and interest) were calculated, and their ratio was determined to provide a measure of cost estimating uncertainty. All costs are stated in 1967 dollars for comparability.

The bottom curve of the figure shows the original average estimates of basic capital cost for each of the years, 1965 through 1970. The middle curve shows the increase in these estimates to mid-1974. The mid-1974 estimates of total costs are plotted as the top curve in the figure; the difference between the middle and top curves is the contribution of price escalation and interest costs.

The estimating uncertainty of the original estimates of basic capital cost, as measured by the ratio of latest estimate to original estimate, increased from 1.34 for plants started in 1965, to 2.00 for those started in 1963, and declined to 1.31 for 1970 plant starts.

These ratios are estimates, except for the plants started in 1965 (all of which were completed), and are subject to further change with time. As of mid-1974, none of the plants started in 1969 and 1970 had been completed, and their ratios of estimates are least certain, as indicated in the figure by the dashed curves.

Estimated costs at the end of 1975 (Ref. 4-7) show further substantial increases. Average estimated mid-1974 total costs at completion had increased by about 20 percent for the 1966 plant starts and about 30 percent for the 1967 plants, most of which had been completed by the end of 1975. Estimates of total cost for plants started in 1968, 1969, and 1970 had increased by about 40, 80, and 40 percent, respectively, and further increases are possible because many of these plants have not been completed.

1967 Dollars/kW_e ^{1/}

Year Orig. Est.	Original Est	less Esc. & Int.	Mid-74 Est.	less Esc. & Int.	Ratio Col. 5 ÷ Col. 3
1	2	3	4	5	6
1965	123	108	206	144	1.34
1966	121	105	260	174	1.66
1967	146	127	354	223	1.76
1968	153	119	403	238	2.00
1969	196	135	371	204	1.51
1970	201	130	335	171	1.31

^{1/} Data adjusted to 1967 dollars. Using BLS wholesale price index, all commodities, for dollar value adjustment

^{2/} Excluded price escalation and interest

Dashed lines, mid-1974 estimates for 1969 and 1970 original estimate: no actual completions. Completed costs will probably be higher than estimated.

Source: F. C. Olds, Power Engineering, August 1974 (Ref. 4-6)

Figure 4-17. Recent Cost Estimating Experience

4.3.3 Estimating Cost Uncertainty

As indicated, initial estimates of nuclear plant costs have tended to be much lower than costs actually incurred. This wide band of uncertainty raises the question: how accurately would surface and subsurface plant costs developed in the ERCDC study, based on initial conceptual design estimates, represent actual relative costs of these plants, if they were actually built?

The question cannot be answered quantitatively with precision. However, the causes of differences, actual versus initial, can be analyzed qualitatively to identify those which may affect subsurface plants to a greater degree than surface plants.

Three categories of factors causing differences between initial estimates and actual costs can be identified for the purposes of discussion:

- a. Apparent differences which result from use of different dollar units of measure, i. e., constant dollars or current dollars
- b. Differences which result from errors in estimating the cost to construct the plant as defined by the initial design and construction plans
- c. Differences which result from changes in design and construction introduced after preparing the initial estimates.

Differences in the first category are mentioned only as a reminder that failure to reduce estimates to common terms for comparison results in confusion and misunderstanding. Clearly, in comparing costs, the same constant-dollar values must be used, and net value, or discounted value, calculations are necessary for comparisons where different schedule times are involved. No differences in estimating uncertainty are involved.

Differences in the second category result from uncertainty in estimating the costs of constructing plants as initially defined by preliminary designs and construction plans. Uncertainties or inaccuracies are inherent in:

- Normalizing experience data to eliminate the cost effects of unusual conditions
- Selection of some parameters to explain cost variation and rejection of others which are difficult to quantify
- Identification of differences between the experience base and the project to be estimated, in terms of parameters in cost relationships
- Adjusting cost relationships to allow for differences in parameters which are not explicitly included.

Where the plant being estimated is very similar to plants in the experience base, empirical estimates of cost relationships apply directly, and uncertainty is confined to inaccuracies in the cost relationships and to moderate variations in parameters not specifically taken into account, e.g., labor morale, labor shortage, materials shortage, or weather. The uncertainty can be allowed for by contingency factors on the order of five to 10 percent.

However, where the plant differs from the experience base in configuration, design criteria, and construction methods and procedures, experience-cost relationships are affected, and uncertainty increases with the degree of change from experience. Changes require that new designs, plans, procedures, and schedules be estimated. The opportunity for errors, ambiguities, and omissions is increased, which may increase material costs (scrap, rework), and decrease labor efficiency (disrupted schedules, idle time, overtime).

The ramifications of changes in the design and construction of a system as complex as a nuclear plant are usually more extensive than can be perceived at the time of initial estimates. Thus, contingency allowances tend not to be adequate, and actual costs of a plant as defined tend to be greater than initial estimates.

Differences in the third category, resulting from significant departures from the defined designs and construction conditions, cannot be realistically or meaningfully accounted for in initial estimates. Major contingencies, such as these, are not included in the normal contingency allowance of initial estimates. Such contingencies include the following, as examples:

- Those estimated to affect surface and subsurface plants approximately equally:
 - Financial difficulties which disrupt schedules.
 - Labor strife. Protracted strikes or slow-downs on site, in equipment suppliers' operations, or in transportation systems.
 - Labor shortage, in general, or shortages of specific skills.
 - Late delivery of equipment or shortage of materials.
- Those expected to have greater effect on subsurface plants:
 - The time required to establish a sufficient regulatory basis.
 - Changes in regulations affecting design and construction.

The cost effects of these major contingencies cannot be estimated for individual plants with useful accuracy. The probability of their occurrence with varying degrees of impact is simply not predictable. Historical data provide little guidance. A large amount of analysis would be required to segregate, even approximately, the "normal" contingencies from the "major" contingencies in the historical data. Even if available, such data would have questionable applicability to an individual future plant.

For this purpose, i. e., to compare costs of alternatives, the major contingencies that affect surface and subsurface installations equally need not be examined in detail.

Those that have a possible differential effect -- the effects of regulatory delays and changes in regulations -- are appropriate to include in comparisons. Regulatory time requirement uncertainties appear to be more likely with subsurface plants, because regulatory agencies have not yet developed the data requirements and procedures for approving the safety and environmental acceptability of these plants. In the course of future development, additional restrictive requirements which affect subsurface plants specifically may be defined.

4.3.3.1 Experience Effects

As discussed above, the principal source of errors and uncertainties in total cost estimates which can be directly attributed to the facility design can be traced to the important factor of inexperience. On this basis, cost estimates for subsurface plants will be somewhat less certain than those for surface plants. The estimates of surface plant costs can be extrapolated from data from a decade of construction experience. Cost relationships derived from these data (i. e., the relationships between labor and material costs and the various jobs which must be performed in plant fabrication and installation) reflect skill and experience in designing, planning, scheduling, and managing surface plant construction.

Subsurface plants require reconfiguration of some subsystems and equipment, and workspace and access to workspace may be restricted. Cost relationships are affected by these differences. New designs, plans, schedules, and installation procedures are accompanied by reduced certainty in estimating. More errors, omissions, and ambiguities can be expected to result in additional cost due to scrap and rework, schedule delays, idle time, and overtime; and the modified construction methods and methods for underground installations which will not have the benefit of improvements that come only with experience.

The magnitude of these effects on costs is difficult to estimate directly; the effects of deficiencies in plans, schedules, and procedures are by nature not known specifically. However, the general effect on cost resulting from experience accumulated in repetitive production, the "experience curve" or "learning curve" effect, provides a method for predicting approximately the uncertainty of estimates for new or first-time jobs, where the level of experience reflected in the basic cost data and relationships can be estimated.

If the basic cost data reflect a level of experience in designing, planning, scheduling, and managing the construction of nuclear plants analogous to that achieved by production of five similar surface plants, then the "first unit" costs of new plant elements can be based on the costs from experience for producing similar elements for the fifth time. Labor costs tend to decline with repetitive production by approximately 20 % (referred to as an "80 % slope") for each doubling of the cumulative quantity produced, as described by the following expression:

$$\begin{aligned} n^{\text{th}} \text{ unit cost} &= (\text{first unit cost})^{\log 0.8 / \log 2} \\ &= (\text{first unit cost})^{-0.322} \end{aligned}$$

By applying this relationship, labor costs for the first unit can be expected to be 68 % greater than costs associated with a level of experience in planning scheduling, managing, etc., achieved after producing five units.

Cost reduction for equipment tends to be less pronounced than the effects on construction costs because a large fraction of the cost of equipment and associated installation procedures is related to costs of materials and standardized fabrication procedures which are insensitive to the production quantities of a particular piece of equipment. Using a 90 % slope for equipment, first unit costs would be greater than the costs associated with producing five units by 28 %.

Table 4-10 shows the estimated distribution of costs for a typical PWR surface nuclear plant from WASH-1230 (Ref. 4-5). (These costs were applicable in 1971. Current costs would be much higher, but the 1971 data is useful for illustrative purposes.) The areas of primary cost impact due to subsurface installation (direct costs and interest and escalation) are noted in the table. Analysis of experience effects on the direct costs is shown in Table 4-11. This table lists the sub-accounts of the baseline cost estimates for the major subsystems of the surface plant which are expected to be significantly affected by subsurface installation, in terms of labor, and equipment and materials costs. The degree of design change required in the A/E concepts developed for underground installations in the subsequent ERCDC investigations has been analyzed in terms of "high", "moderate", or "low" degree of change. The high degree of change is estimated to result from "first time" or "first unit" costs associated with the A/E designs. As a consequence of designer inexperience, as previously noted, labor costs for the first unit are expected to rise by 68 %, equipment and materials by 28 %. "Moderate" changes were estimated to increase costs by half as much. "Low" change would be associated with negligible increases above the experience cost base.

To estimate the actual cost uncertainty associated with the design of an underground facility, it would be necessary to develop cost estimates (in the degree of detail shown in Table 4-11) for the complete unit. However, a first-order estimate of the impact of design related uncertainties in costs for the overall facility can be made from an analysis of the baseline surface

Table 4-10. Distribution of Surface Nuclear Plant Capital Costs

<u>Item No.</u>	<u>Account No.</u>		<u>Capital, \$M, 1971</u>	<u>Effect of Installing Plant Underground</u>
	20	Land & Land Rights	1.0	
	21	Structures & Improvements	33.5	
	22	Reactor Plant Equipment	54.5	
	23	Turbine Plant Equipment	62.3	
	24	Electric Plant Equipment	14.5	
	25	Misc. Plant Equipment	4.2	
1	Total Direct		170	Increase
2	Contingency		13	
3	<u>Undistributed</u>		42	Same
		Professional Services	24	
		Other Indirect	18	
4	<u>Basic Plant Cost</u>		225	
5	<u>Interest and Escalation</u>		<u>114</u>	Increase
6	<u>Total Plant Cost</u>		339	Increase

Source: WASH-1230, January 1971 experience data.

Table 4-11. Impact of Subsurface Installation on Plant Designs

<u>Account</u>		<u>Direct Cost \$M, 1971</u>			<u>Design Impact</u>	
		<u>Labor</u>	<u>Equipt & Mat'l</u>	<u>Total</u>	<u>Rock Cavity</u>	<u>Berm</u>
21	<u>Struct. & Improvemts</u>	<u>21.2</u>	<u>12.3</u>	<u>33.5</u>		
211	Yardwork	1.4	1.0	2.4	high	high
212	Containment Bldg.	8.8	5.0	13.8		
	(Substructure)	(0.9)	(0.5)	(1.4)	high	high
	(Superstructure)	(7.2)	(3.2)	(10.4)	high	high
213	Turbine Room	2.7	2.1	4.8	high*	high*
214	Intake & Discharge	1.8	0.8	2.6	high**	high**
215	Primary Aux. Bldg.	3.0	1.2	4.2	high	high
217	Fuel Handling Bldg.	0.9	0.5	1.4	high	high
218 A	Control Rm Bldg.	0.9	0.4	1.3	low	low
218 B	Diesel Grn. Bldg.	0.4	0.2	0.6	low	low
218 C	Admin. Bldg.	0.4	0.5	0.9	low	low
218 D	Service Bldg.	0.5	0.5	1.0	low	low
218 E	Fan Rm. Bldg.	0.2	0.1	0.3	high	high
218 F	Aux. Feed Pump	0.1	0.1	0.2	moderate	mod
22	<u>Reactor Plant</u>	<u>9.4</u>	<u>45.1</u>	<u>54.5</u>		
221	Reactor Equipt.	0.7	13.9	14.6	low	low
222	Reactor Coolant System	2.4	20.9	23.3		
	(Piping)	(0.9)	(2.1)	(3.0)	high	high
223	Safeguards Cooling System	1.5	1.7	3.2	high	mod
224	Radioactive Waste	0.4	0.6	1.0	moderate	mod
225	Nuclear Fuel Handling	0.6	0.9	1.5	high	mod
226	Other Reactor Pl. Equipt.	2.7	2.8	5.5	high	mod
227	Instrum. & Control	1.1	4.2	5.3	moderate	mod
23	<u>Turbine Plant Equipt</u>	<u>15.9</u>	<u>46.4</u>	<u>62.3</u>		
231	Turbine Gen.	2.8	30.7	33.4		
	(T. G. Support)	(0.6)	(0.3)	(0.9)	moderate	mod
232	Circulating Water	2.3	2.5	4.8		
	(Circ. Water Ducts)	(0.5)	(0.3)	(0.8)	low	mod
233	Condensing System	2.8	4.3	7.1		
	(Condensers)	(0.8)	(2.9)	(3.7)	low	mod
234	Feedwater System	3.4	4.6	8.0		
	(Feedwater Piping)	(1.8)	(1.1)	(2.9)	high	high
235	Other Turbine Eq.	4.0	3.3	7.3		
	(Main Vapor Piping)	(2.4)	(1.5)	(3.9)	high	high
236	Instrum. & Control	0.6	1.0	1.6		
	(Instr., Piping, Wiring)	(0.5)	(0.2)	(0.7)	low	mod
24	<u>Electric Plant Equipt.</u>	<u>6.2</u>	<u>8.3</u>	<u>14.5</u>		
	Switchgear	0.2	1.1	1.3	low	low
	Station Service Equipt.	0.8	3.8	4.6	low	low
	Switchboards	0.1	0.5	0.6	low	low
	Protective Equipt.	0.2	0.1	0.3	low	low
	Electr. Struct. & Wiring	1.2	0.4	1.6	moderate	mod
	Power & Control Wiring	3.8	2.4	6.2	moderate	mod

* Applies if item is placed below grade

** Applies if item 213 is placed below grade

plant cost estimates alone. Results of an analysis of this type are summarized in Table 4-12. The net effect of design inexperience, as influenced by translation of surface plant experience to underground facility design, is shown to be cost uncertainty of \$28.1 million, or roughly 15% of the basic surface plant direct cost of \$225 million. *

Thus, if the surface plant designs and construction procedures are changed to the degree estimated, a direct 15 % increase in cost can be expected as a result of uncertainties in costs estimated, using the surface plant experience base in the process of design. The figure "15 %" is a rough approximation, as is evident from the method of estimating design change impact in terms of "high", "moderate", and "low", and further, in estimating the current experience level of architect-engineer-constructors to be equivalent to that attained in constructing five very similar surface plants.

Architect-engineer estimates can allow for experience effects by appropriate reductions in labor productivity, for example. However, the tendency in estimating is to apply cost relationships from experience to plant subsystem elements and to understate cost effects due to influences such as "inexperience" effects which are difficult to segregate from experience data and are thus uncertain.

The 15 % increase is believed to be representative of the band of uncertainty which might be additive to A/E estimates. Part of this amount may be included in estimates, as noted, through adjustment of labor productivity factors from experience data. However, such adjustments will probably be offset by additional uncertainty not calculated above: the inexperience effect on labor costs of "moderate" and "low" impact items would be due to new methods and procedures required to install equipment subsurface with restricted access and workspace.

* The \$225 million basic direct cost estimate was developed on the basis of Jan 1971 cost experience data (Ref 4-5). More recent estimates of basic direct costs have shown escalations to about \$600 million for similarly sized plants, based upon 1976 cost data.

Table 4-12. Effect of Experience on Direct Costs of Subsurface Plant in Rock Cavities

Cost, \$M, 1971

<u>Account</u>	Labor Costs			Equip & Matl Costs		
	Ref Cost (1)	Factor (2)	Added Cost (4)	Ref Cost (1)	Factor (3)	Added Cost (4)
<u>High Impact Items</u>						
21 Struct & Improvements	18.6	0.67	12.5	10.8	0.28	3.0
22 Reactor Pl. Equipment	5.7	0.67	3.8	7.5	0.28	2.1
23 Turbine Pl. Equipment	4.2	0.67	2.8	2.6	0.28	0.7
24 Electric Pl. Equipment	0	0.67	0	0	0.28	0
TOTAL	28.5		19.1	20.9		5.8
<u>Moderate Impact</u>						
21 Struct & Improvements	0.1	0.33	0.0	0.1	0.14	0.0
22 Reactor Pl. Equipment	1.5	0.33	0.5	4.8	0.14	0.7
23 Turbine Pl. Equipment	0	0.33	0	0	0.14	0
24 Electric Pl. Equipment	5.0	0.33	1.6	2.8	0.14	0.4
TOTAL	6.6		2.1	7.7		1.1
<u>Low Impact</u>						
20 Land & Improvements	0	0		1.0	0	
21 Struct & Improvements	2.5	0		1.4	0	
22 Reactor Pl. Equipment	0.1	0		14.3	0	
23 Turbine Pl. Equipment	11.7	0		43.8	0	
24 Electric Pl. Equipment	1.2	0		5.5	0	
25 Misc Pl. Equipment	2.1	0		2.1	0	
TOTAL	17.6		0	68.1		0
TOTAL <u>DIRECT</u>	52.7		21.2	96.7		6.9

Reference Plant Basic Cost (Ref. Table 4-10, Item No. 4) \$225M
 Estimate Uncertainty Cost \$28.1M
 Estimated Direct Cost Uncertainty of Subsurface Plant 12.5%

- (1) WASH-1230, January 1971 experience data.
- (2) Fractional increase in cost due to "inexperience". For high design impact items, factor is first unit cost relative to fifth unit cost assuming 80% log-linear unit experience curve. For moderate impact items, assume half as much increase.
- (3) Same as (2) except 90% curve.
- (4) Added Cost: "experience curve" effect on reference costs due to design changes.

4.3.3.2 Estimated Total Uncertainty

To judge from the historical record, costs of nuclear plants can be expected to exceed initial cost estimates by 30 % to 100 % in terms of constant dollars. The uncertainty in the historical data can be attributed in large part to a lack of nuclear construction experience on which to base estimates for the earlier plants and to continual changes in requirements for design and construction of the later plants. Requirements are expected to continue to be changed. A recent article, Ref. 4-5, states that 36 new regulatory requirements having significant impact on design were issued during 1976. Further changes can be expected as the generalized requirements for "clean" air and "safe" nuclear plants are transformed into definitive regulations. The magnitude of cost uncertainty due to future changes in requirements is not known. However, uncertainty similar to that demonstrated by historical data, 30 to 100 %, might be expected, and it should affect costs of surface and subsurface plants approximately equally.

Uncertainty due to experience effects related to changes required for subsurface installation is approximately 15 %. Thus, if the uncertainty associated with surface plants is estimated to be 30 to 100 %, that associated with subsurface plants may be as high as 45 to 115 % above the A/E cost estimates to be developed in the subsequent portion of the ERCDC program unless the inexperience factor is included in contingency estimates directly.

The information shown in Table 4-12 can be interpreted as general guidance on the selection of levels of A/E design detail appropriate to the desire to reduce cost estimate uncertainty. For example, the tabular data in the right-hand column shows that both Structures and Improvements and Reactor Plant Equipment dominate the uncertainty estimates for equipment and material costs. This general observation is not altogether surprising. And, if considered alone, the data does not provide any real guidance to the question of where emphasis should be placed to achieve the goals of reduced cost uncertainty. The additional information necessary to make such decisions is contained in the more detailed breakdown of system cost accounts shown in Table 4-11.

Here it can be seen that in the case of the previous example, that the containment building superstructure and substructure, the reactor plant coolant system piping, and the safeguards cooling system make the principal contributions to the uncertainty. Other subsystems with high design impacts will also contribute, but in descending order of importance according to their relative baseline costs.

Thus, it is possible to predict in advance where attention to design should be directed (in terms of increasing the level of engineering detail) in order to achieve goals which might be set to reduce the value of cost uncertainty in the facility design. In order to methodically reduce uncertainty, it will be necessary to develop the details of the conceptual engineering design in rough proportion to the relative baseline costs and impacts specified in Table 4-11. While the results of applying such criteria for establishing the priorities for various aspects of design may not be absolutely correct, they are believed to be adequate guidelines for achieving a quick overview of the appropriateness of design efforts.

4.4 ANALYSIS OF POTENTIAL UNDERGROUND PLANT SYSTEM CONFIGURATION

The structures in a surface facility which house the critical plant systems were briefly described in Section 4.2 along with the interfaces which exist between subsystems. This material has been presented in order to lay the foundation for selection of components for location underground. This section presents the methodology for identifying the available options for specifying locations of major plant systems, examining the implications of each option, and finally making recommendations for surface or underground locations for the major plant systems. A guideline used in making those determinations was that the underground plant should be designed for minimum facility cost consistent with safety objectives.

4.4.1 Selection of Undergrounded Components

One way to study the arrangement of the components of a nuclear power plant is to develop a functional description of the plant and all of its systems; then to use either a tree or network diagram to identify and analyze options for undergrounding. Such a functional description is shown in Table 4-13, in abbreviated tabular form. Using this information, the network diagram shown in Figure 4-18 was constructed. The analysis included an examination of all possible groupings of plant subsystems, considering only those which permit the plant to function properly.

These analyses identified seven basic options for the systems to be undergrounded. These are listed in Table 4-14 and shown in Figures 4-19a through 4-19f. Ideally, each of the options should be examined for both LWR concepts, and a final selection made on the basis of economics and safety. However, this procedure is time-consuming and probably not necessary. It appears likely that a more critical examination may permit a marked reduction in the number of options requiring careful consideration. The options shown in Table 4-14 have been arranged

Table 4-13. Nuclear Power Plant Functional System Relationships

Description	Function	Relationship
1 - Structures & Improvements		
1. Yardwork	1. Roads, Rails for Access and Permanent Use	1. Road to Rad Waste, T-G and Contain. Building, Rail to T-G and Contain. Bldg.
2. Main Power Station Building	2. N/A	2. N/A
3. Containment Building	3. Control as the Release of Reactor Fluid/Cases to a Building	3. Encloses the entire Reactor Steam Generating and some Safety Systems.
4. Turbine Room & Heater Bay	4. Encloses the Turbine-Generator System	4. Physically supports the System Components and gives Shelter to T-G.
5. Intake & Discharge Structures	5. Provides Head Works and Foundations for Cooling Water Components	5. Interface for Offsite Cooling Water Supply and Return
6. Primary Auxiliary Building	6. N/A	6. N/A
7. Radwaste Building	7. Encloses the Rad Waste to Processing and Storage Equipment	7. Reactor Coolant Loop and Spent Fuel Pit Find Loop Connections
8. Fuel Handling Building	8. Storage of Used Fuel	8. Inface with Reactor
9. Control Room Building	9. Monitors and Provides Central Control of the Plant	9. Electrical, Pneumatic and Power Connection to All Areas of Plant
10. Diesel-Generator Building	10. Encloses the Diesel Generator	10. Interconnects with Main Power Distribution
11. Administration Building	11. Administration and Recruiting Housing	11. None Functional
12. Service Building	12. Enclosure for Maintenance and Support Equipment and Personnel	12. None Directly During Operating
13. Fan Room Building	13. N/A	13. N/A
14. Auxiliary Feed Pump Enclosure	14. N/A	14. N/A
15. Stacks	15. Discharge Radioactive Effluents	15. Interface with Containment and Rad Waste Building
2 - Reactor Plant Equipment		
1. Reactor Equipment	1. Contains, Supports the Reactor Core and Coolant Fluid	1. Within the Containment Building. Connection to T-G Bldg.
2. Reactor Coolant System	2. Pumps Water Through the Reactor and Generator Steam	2. Mechanically with Reactor Equipment, Electrically with Power Supply
3. Safeguards Cooling Systems	3. Circulates Water to Remove Heat When Reactor is Shutdown	3. Reactor Equipment Through Containment Building
4. Radioactive Waste Systems	4. Liquid, Gaseous and Solid Waste Handling System	4. With Reactor Through Containment Building
5. Nuclear Fuel Handling Systems	5. Handling and Storage of Fuel	5. Next to Containment Building and Integral with Reactor Contain. Core Rad Shield
6. Other Reactor Plant Equipment	6. Coolant Cleanup, Building Cooling	6. External to Reactor Containment
7. Instruments and Controls	7. Monitors and Operates Reactor Plant	7. Both Internal and External to Reactor Containment, Interface W/CR
3 - Turbine Plant Equipment		
1. Turbine Generator Equipment	1. Converts Steam Energy to Electric Energy	1. Connects to Reactor Steam Discharge and Water Supply, Includes Foundation
2. Condenser Water System	2. Pumps, Piping and Equipment for Supplying Cooling Water	2. Connects to T-G Condenser and Cooling Water Supply
3. Condensing Systems	3. Condenses T-G Discharge Steam	3. Connects Directly to T-G
4. Feedwater Systems	4. Heats and Pumps Feedwater	4. Connects to Reactor and T-G Condenser
5. Other Turbine Plant Equipment	5. Reactor to T-G Steam Lines Connection and Cooling Pumps	5. Internal to T-G Building
6. Instruments and Controls	6. Monitors and Operates Turbine-Generator Plant	6. Interface with CR
4 - Electric Plant Equipment		
1. Switchgear	1. Control and Monitoring of Generator Output	1. Connects to Generator Power Supply
2. Station Service Equipment	2. AC/DC Emergency Power	2. Supplies Emer. Power to Plant
3. Switchboards	3. Main Control and Relay Boards	3. Located in Control Room
4. Protective Equipment	4. Fire, Lighting Cathodic	4. Critical Area of U/G Steel, Fire Areas
5. Elect. Struct. and Wiring Cont.	5. Cable Ways, Conduits	5. Connects to All Plant Areas
6. Power and Control Wiring	5. Main Gen. Bus.	6. T-G Building

Figure 4-18. Network Diagram Representative of a Nuclear Power Reactor System

Table 4-14. Underground Berm Configuration Options

<u>Concept^a</u>	<u>Plant Elements Underground</u>
1. Reactor	Reactor Pressure Vessel (RPV)
2. Reactor Containment	Nuclear Steam Supply Systems (NSSS)
3. Reactor System	NSSS, Safeguards Cooling, Other Reactor Plant
4. Reactor Plant	NSSS, Safeguards Cooling, Radwaste, Fuel Handling, Other Reactor Plant
5. Reactor System & TG	NSSS, Safeguards Cooling, Other Reactor Plant and Turbine-Generator (TG)
6. Reactor Plant & TG	NSSS, Safeguards Cooling, Radwaste, Fuel Handling, Other Reactor Plant and Turbine-Generator
7. Total Plant Underground	Same as Above Plus Control Center and Emergency Power

^aOrdered by the fraction of the total plant placed underground.

Figure 4-19a. Concept Option 1, Reactor Pressure Vessel Undergrounding

Figure 19b. Concept Option 2, Reactor Containment Undergrounding

Figure 4-19c. Concept Option 3, Reactor System Undergrounding

Figure 4-19d. Concept Option 4, Reactor Plant Undergrounding

Figure 4-19e. Concept Option 5, Reactor System - Turbine Generator Undergrounding

Figure 4-19f. Concept Option 6, Reactor Plant & Turbine Generator

in order of increasing fraction of the plant placed underground beginning with only the reactor underground to the final option where the whole plant is underground.

It appears that undergrounding will generally increase the subsystem interconnection lengths, a critical-cost factor. For plants with equipment installed in rock cavities, it will be necessary to separate the cavities to achieve the required cavern structural integrity. To ensure the soundness of the cavity structure, the thickness of the walls must be related to the size of the cavities and the burial depth. Depending upon the characteristics of the rock, the wall thickness between adjacent cavities may be equal to one or more cavity-spans. Under these conditions, for a cavity span of 100 feet, which Table 4-3 shows to be the approximate requirement, the separation distance between systems can become a few hundred feet and significantly increase plant costs. Greatly increased separation distances between systems would also occur if some systems were placed at grade and others below. Increases in costs of interconnections would then be determined by burial depth, surface topography and plant access design.

For berm facilities, it may be possible to place the structures housing the several subsystems close together and thereby achieve a lower interconnection cost than the rock cavity concepts permit. However, the additional cost of increased structural strength to support the overburden on buildings may offset these benefits. To explore these questions and provide a measure of quantitative information for a rock cavity facility, the difference in costs of subsystems and interconnections was estimated for each of the underground options of Table 4-14. The cost data in Table 4-15 summarize the results. Figures 4-20 through 4-24 illustrate the assumed equipment relationships for the seven undergrounding concepts analyzed. Beneath each diagram is the cost account data taken from WASH-1230 (Ref. 5-3) for that portion of the system which was

Table 4-15. Differential Costs for Optional Underground Concepts

ACCT No.	ACCOUNT DESCRIPTIONS	SURFACE PLANT DIRECT COST (M\$)	DIFFERENTIAL COSTS FOR UNDERGROUND CONCEPTS (M\$)						
			1	2	3	4	5	6	7
20	LAND AND LAND RIGHTS	1.0	0	0	0	0	0	0	0
	SUBTOTAL		0	0	0	0	0	0	0
21	STRUCTURES AND IMPROVEMENTS	33.5							
	CONTAINMENT BUILDING		0	-11.8	-12.7 ^d	-12.7 ^d	-13.1 ^e	-13.1 ^e	-13.1 ^e
	U/G CONTAINMENT		+ 1.4 ^a	+ 3.3	+ 3.3	+ 3.3	+ 3.3	+ 3.3	+ 3.3
	AUXILIARY BLDG		0	0	+ 1.2	+ 0.8 ^j	+ 0.8	+ 1.8	+ 2.1 ^k
	T-G EXCAVATION		0	0	0	0	+ 4.5	+ 4.5	+ 4.5
	SUBTOTAL		+ 1.4	- 8.5	- 8.2	- 7.6	- 4.5	- 3.5	- 3.2
22	REACTOR PLANT EQUIPMENT	54.5							
	REACTOR COOLANT SYSTEM		+18.6 ^b	0	0	0	0	0	0
	COOLANT SYSTEM SHAFT		+ 0.2	+ 0.1	+ 0.1	+ 0.1	+ 0.1	+ 0.1	+ 0.1
	SAFEGUARDS		+ 4.8	+ 4.8	0	0	0	0	0
	RAD WASTE AND FUEL TRANS		+ 1.3 ^c	+ 1.3 ^c	+ 1.3 ^c	+ 0.4 ^b	+ 1.3 ^c	+ 0.4 ^b	+ 0.4 ^b
	SUBTOTAL		+25.0	+ 6.2	+ 1.4	+ 0.5	+ 1.4	+ 1.4	+ 1.4
23	TURBINE PLANT EQUIPMENT	62.3							
	FEED WATER AND STEAM LINES		0	+20.4 ^b	+20.4	+20.4	+ 6.8 ^g	+ 6.8 ^g	+ 6.8 ^g
	STEAM SHAFT		0	+ 0.1	+ 0.1	+ 0.1	0	0	0
	FOUNDATION MAT		0	0	0	0	+ 0.2	+ 0.2	+ 0.2
	SUBTOTAL		0	+20.5	+20.5	+20.5	+ 7.0	+ 7.0	+ 7.0
24	ELECTRIC PLANT EQUIPMENT	14.5							
	POWER CABLES		0	0	0	0	+ 1.8 ^b	+ 1.8 ^b	+ 1.8 ^b
	SHAFT		0	0	0	0	+ 0.1	+ 0.1	+ 0.1
	SUBTOTAL		0	0	0	0	+ 1.9	+ 1.9	+ 1.9
25	MISCELLANEOUS PLANT EQUIPMENT	4.2							
	SUBTOTAL		0	0	0	0	0	0	0
	TOTALS	170 ^l	+26.4	+18.3	+13.7	+12.4	+ 5.8	+ 6.8	+ 7.1

a - Assumes a 70 ft D x 120 ft H cylinder at \$20/yd, 1000 MW electric

b - Assumes a 300 ft depth

c - 1.0M for radwaste, 0.3M for fuel transfer

d - Includes 0.9M credit for auxiliary bldg

e - Includes 0.4M credit for T-G substructure and excavation

f - Includes 4.5M for T-G chamber

g - Assumes 100 ft separation from NSSS chamber

h - Assumes 100 ft separation from NSSS chamber

j - Includes 0.6M for fuel transfer

k - Includes 0.3M for control center

l - Direct cost estimate from WASH-1230, Jan 1971; total plant cost of 339 M\$ if contingency, professional services, other direct, interest and escalation are included

Figure 4-20. Reactor Shutdown Cooling Systems

Figure 4-21. Safeguards Cooling System

Figure 4-22. Reactor Primary Coolant System

Figure 4-23. Liquid Rad-Waste System

Figure 4-24. Main Steam and Feed Water System

changed relative to a surface-sited facility. Each table shows the approximate cost (in 1971 dollars) of the equipment identified by the heavily marked portion of the figures. This was done for the six major equipment cost accounts. The bottom line of Table 4-15 shows the overall cost impact. It is clear that the options displaying the greatest impact are 1, 2, 3, and 4, all of which have extensive remote interface characteristics. Of these, option 1 provides the most severe impact because of the increase in cost of interconnections to remote steam generation equipment and the increase in cost of the underground containment which includes the reactor excavation. In this concept the reactor pressure vessel is underground but the steam generators and pumps, normally located close to the reactor pressure vessel (RPV), remain at the surface. However, it appears that the steam generator and interconnecting piping would still have to be inclosed in a leak-proof containment structure, much like a surface design. The option showing the least relative cost increase is Number 5.

The relative cost differential numbers shown on Table 4-15 are intended to be used as typical rather than actual current prices for nuclear plant equipment. Moreover, the information on the table should not be interpreted as an absolute comparison of cost differences, but rather as an indication that significant cost differences can result from different underground plant configurations.

In summary, it appears that the high cost of system interconnections can offset any cost advantages of a plant constructed partly above and beneath the surface, particularly if the surface-to-subsurface interconnection distances are great. Therefore, it is concluded that the least cost concept is not necessarily one which minimizes underground components.

4.5 EVALUATION OF CONFIGURATION OPTIONS

It is the objective of this section to provide a preliminary evaluation of the several underground siting concepts identified previously. The material presented here contributed to the preparation of specifications for the conceptual design studies to be performed in the second phase of the program.

4.5.1 Evaluation Criteria

The issues which were considered in the formulation phase of the criteria were restricted to general topics of a technical nature; only a few nontechnical issues (e.g., aesthetics, water availability, land use) that could significantly influence future decisions of nuclear plant design and siting were included. The criteria were selected with the objective of exploring the feasibility and economics of underground siting. Consequently, the selection was conducted with a conscious effort not to be so definitive that the criteria themselves might discourage or restrain the exercise of ingenuity and creativity in the development of engineering designs. The issues of containment configuration and design that were identified for the preliminary design studies are listed in Table 4-16. Only the issues identified with an asterisk were considered within the scope of this study.

4.5.2 Comparative Evaluation of Options

As discussed in Section 4.1, the basic concepts to be evaluated are:

<u>Soil-Covered</u>	<u>Rock Cavity</u>
Cover Only	Hillside Entry
Cut-and-Cover	Vertical Entry
Balanced Cut-and-Cover	
Soil-Rock	

Table 4-16. Principal Concept Evaluation Issues

1 - Plant Configuration and Layout

- * a - Structures and Systems to be Buried
- b - Depth of Burial
- c - Seismic Analysis
- * d - Reactor System
- * e - Plant Access

2 - Siting and Geology

- * a - Rock Geology
- * b - Soil Geology
- * c - Water Table
- d - Site Access
- e - Surface Topography
- f - Meteorological Attributes
- g - Cooling Water Supply

3 - Economics and Schedules for Construction

- * a - Excavation Costs
- b - Plant Costs
- c - Operation Costs
(Fuel, Personnel and Maintenance)

4 - Social and Environmental

- * a - Aesthetics
- b - Public Hazard
- c - Utilization of Resources

*Asterisked parameters indicate aspects included within scope of this study.

An overall evaluation of these concepts has been conducted relative to the issues identified in Table 4-16. Based on information presented in the previous sections, qualitative estimates of optimal locations (either underground or at the surface) for structures and systems for the above concepts are shown in Table 4-17. On the basis of the table, it appears that the balanced cut-and-cover berm-contained concepts and horizontal entry mined-rock concepts could most completely satisfy the objectives of buried or underground plant construction by allowing more of the plant systems to be placed underground. The decisions for "optimal" structure/system location were predicated on a desire to minimize interconnection lengths and corresponding costs, reduce access design and isolation complexity, and reduce the threat of external hazards such as flood, slides, etc. The exercise of judgement, based on experience accumulated to date in underground plant designs and studies, was also included in the qualitative decision process.

A summary of the comparative characteristics for the several concepts described in the previous section is shown in Tables 4-18 and 4-19.

To summarize, on the basis of the information in Table 4-17 for "optimal" locations of Structures and Systems and the Summary of Comparative Concept Design Characteristics (Tables 4-18 and 4-19), it appears that the balanced cut-and-cover or soil-rock concepts are the most attractive configurations for berm designs. Similarly, the hillside entry design appears to be slightly superior to the vertical entry design for siting in rock. These configurations are thought to encompass the best design features of the two basic approaches to underground plant-construction, while taking advantage of favorable layout and plant arrangement geometries. It is important to observe that the selections proposed herein are tentative and could easily be changed or modified by subsequent discussions with nuclear industry plant designers and engineers.

Table 4-17. Qualitative Estimates of Optimal
Structure System Locations

Structure System / Concept	Cover	Cut-&- Cover	Balanced Cut-&- Cover	Soil- Rock	Hillside Entry	Vertical Entry
Turbine Generator	U	S	S	S	U	S
NSSS Containment	U	U	U	U	U	U
Rad Waste	U	S	U	S	U	U
Fuel Storage	U	S	U	U	U	U
Diesel Generator	S	S	U	U	U	S
Main Transformer	S	S	S	S	U	S
Control Center*						

U - Indicates location to be below grade or covered.

S - Indicates location to be at grade.

* - Location optional.

Table 4-18. Comparative Characteristics
of Berm Concepts

	Cover Only	Cut-&- Cover	Balanced Cut-&- Cover	Soil-Rock
Cut & Fill Volume ^a , yd ³	11 E6	4.5 E6	1.6 E6	0.6 E6
Fill Volume ^a , yd ³	--	--	0.9 E6	0.4 E6
Aesthetic Impact	Large	Small	Moderate	Moderate
Siting Area Req. ^a , ac	40	15	8.6	6.5
Dewatering	NA	Difficult	Difficult	Difficult
Cut & Fill Cost ^a , \$M	7	6.8	3	1.6
Site Geology Description	Sand, gravel, & silt	Sand, gravel, & silt	Sand, gravel, & silt	Sand, gravel & silt; rock at 50 ft plus

^aContainment structure only.

Table 4-19. Comparative Characteristics
of Mined-Rock Concepts

	Hillside Entry	Vertical Entry
Excavation Volume ^a , yd ³	0.6 E6	0.4 E6
Aesthetic Impact	Small	Small
Surface Area Req. ^a , ac	10	25
Cooling Water Head	Low	High
Siting Availability Potential	High	Moderate
Excavation Cost ^a , \$M	8	12
Access	Relatively easy	Relatively difficult

^a Includes Containment, Fuel Storage and Waste Storage Building

REFERENCES

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5.0 EVALUATION OF CONCEPTS FOR AUGMENTING SAFETY OF SURFACE NUCLEAR PLANTS

Since underground siting is expected to involve engineering modifications with resultant cost penalties over conventional surface plants, it was deemed prudent to explore alternative methods of increasing the safety of above ground reactors in the event of a Class-9 accident. If a decision is to be reached relevant to the necessity for constructing nuclear power plants underground, the feasibility and cost of surface-sited alternative concepts should be understood. The following section presents an analysis of methods to control and/or eliminate potential fission product releases from surface facilities, in the event of a Class-9 accident, based largely upon a synthesis of concepts developed and presented in the available literature.

5.1 APPROACH AND GROUND RULES

A literature survey was performed of published concepts relevant to improving the safety of surface-sited Light-Water Reactors (LWRs) with respect to core-melt accidents. A screening of the material produced by the literature search revealed that, for the most part, designs for aboveground systems had not been developed much beyond the conceptual level for coping with a Class-9 accident (i. e., with some exceptions, little serious engineering design had been performed in connection with the concepts). Further, the literature generally addressed remedies at the subsystem level (i. e., most concepts were designed to control or mitigate a specific phenomenon, with insufficient capability to provide complete control of a core-melt accident). Therefore, for purposes of this study, it was considered appropriate to combine or modify the published concepts, in order to define conceptual safety systems which could provide complete control of a Class-9 accident.

Technical goals were adopted to assist in a first-order screening of concepts from the literature. To meet these goals, the concepts were required to have:

- Promise of significantly improved safety for above-ground reactors; that is, modifications promising only modest safety improvements, such as longer times before containment failures, were screened out as being insufficiently attractive for the objective.
- Promise of complete control of core-melt accidents; as shown later, complete control does not necessarily imply complete containment.
- Simple solutions; simplicity, in general, is expected to be highly correlated with reliability and high confidence.
- Passive operational mechanisms; active systems are generally subject to the criticism that there is a finite probability that they will fail to perform as intended because of mechanical, electrical, or human failure.

Evaluation criteria were developed for a qualitative ranking of the concepts considered. Comparative evaluations were made for each criterion; however, the concepts were not considered to be sufficiently mature to warrant an overall ranking.

Areas of technical uncertainty were identified for each system concept with the objective of identifying areas where further work might improve confidence in the technical feasibility of the concept.

It was not possible to assess the cost impact of implementing the concepts considered in this study. The engineering was generally not sufficiently mature, and adequate research to establish technical feasibility had not always been performed. However, the authors of these concepts were usually quite cognizant of the importance of cost, and did not make proposals that were obviously outrageously expensive. However, it is fair to note that inventors are generally optimistic in their cost estimates. With these provisos noted, it was observed that the safety concepts put forth were judged by their authors to impact total plant construction cost by no more than 10 per cent.

5.2 LWR CORE MELT ACCIDENT SEQUENCE

The literature reveals a rather widespread acceptance of the general sequence of events associated with a hypothesized massive core-melt. The usual initiating event is taken to be a loss-of-coolant accident (LOCA), although other initiating events, such as total electrical power failure or acts of sabotage, are also potential sources of Class-9 accidents.

Table 5-1 shows a possible sequence of events associated with a melting core together with phenomena associated with each step. The material presented in Table 5-1 summarizes the events associated with the core-melt accident. A more complete description has been presented in Section 3.1.

Table 5-1a. Possible Sequence for Massive LWR Core Melt

EVENTS (In Approximate Chronological Order)

Blowdown of Primary Coolant

ECCS Injection

Boiling in Pressure Vessel

Core Overheats

Steam generation

Zircalloy oxidation

H₂ gas generation

H₂ explosions possible

Core Melts

Volatile fission products released and dispersed

Core Slumps into Bottom of Reactor Pressure Vessel

Steam explosions possible

Core Meltthrough Bottom of Pressure Vessel

Steam explosions possible

Steam generation from sump fluids

Core Attacks Concrete Floor

Steam generation

CO₂ gas release

Table 5-1b. Potential Containment Failure Mechanisms

Containment Overpressure by

Steam + decay heat

H_2

H_2O plus CO_2 from concrete

Energy Within Containment Building (Decay Heat) Fails

Containment Walls Thermally

Core Penetrates Containment

It is generally agreed that a massive core-melt will eventually lead to meltthrough of the reactor pressure vessel and core penetration through the outer containment building. Overpressurization of the containment with resulting uncontrolled venting is a probable consequence if some mechanism for pressure relief is not provided. It is not certain, however, that (a) steam or hydrogen-air explosions will occur, (b) massive venting will result from the core entering the earth, or (c) fission product heating will in itself fail the containment walls. Most sources, however, agree that the possibility of steam explosions cannot be neglected when the core-melts through the pressure vessel when water has collected beneath the reactor as a result of blowdown and release of the fluid in the Emergency Core Cooling System (ECCS). Similarly, the probability of unrestricted venting of radioactive fission products from the containment increases as the containment overpressure increases prior to meltthrough.

Containment Failure

From the above discussion and Table 5-1, it is clear that a number of phenomena can individually or collectively lead to containment violation, and each must be controlled in order to contain the meltdown accident and prevent the release of fission products. For convenience, the classes of phenomena have been grouped as follows:

- a. Steam Explosions
- b. Vapor-Generated Overpressurization
- c. Core Penetration of Containment
- d. Long-Time Heat Generation.

Steam explosions can violate containment by propelling missiles against the walls, by shock waves, or by incremental increases in the containment overpressure. The explosions may occur whenever the molten core (or portions of it) falls into water well below its vapor saturation temperature. It should be observed, however, that the steam explosion phenomenon is not well understood, and its probability of occurrence is widely debated.

In the event of a major core meltdown, vapor-generated overpressurization is considered to be highly probable for conventional containment buildings because they are sized primarily to contain the steam generated by the blowdown process from a LOCA. When the containment pressure is increased by the hydrogen from the zirconium-steam reaction, sufficient pressure is probably produced to exceed these specifications. Moreover, additional steam and CO_2 from the meltdown produce comparable quantities of gas which may further exacerbate an already difficult problem.

Penetration of the foundation basemat of the containment occurs after the core-melts through the reactor pressure vessel. Then the molten core, driven by gravity, with thermal energy supplied by fission product decay heat, will inexorably melt through customary thicknesses of steel and concrete in the foundation of the containment building.

Long-time heat generation occurs not only as a result of energy released from the molten core, but also by direct heating of the containment atmosphere due to decay of vaporized volatile fission products, which account for a high proportion (up to 30 per cent by some estimates) of the decay heat. Since the heat capacity of the air in the containment is wholly inadequate to forestall a rapid rise in its temperature, there are grounds for concern that the temperature of the containment walls will rise beyond the point which the material can withstand. Melting of steel and spalling and decomposition of concrete are the primary mechanisms for the thermal failure of the containment. Excessive temperatures may also be expected to degrade and destroy containment isolation seals and permit leakage into other plant areas.

5.3 ALTERNATIVES

5.3.1 Siting Options

The principal alternative siting options appear to be remote land-based and offshore (ocean) siting.

Remote land-based siting presupposes that sufficiently remote sites are available to prevent dangerous exposures to humans in the event of a Class-9 accident. The American Physical Society review of LWR safety (Ref. 5-1) states that ". . . it may be quite difficult to site a reactor so remotely as to reduce the average expected population dose from a major release of radioactivity by as much as an order of magnitude." Since even an order of magnitude falls short of the objective of this study, remote siting was judged to have limited potential for providing significant safety benefits in the event of a severe accident.

Offshore plants of current design do not appear to offer significant safety advantages against meltdowns. If a massive melt occurred in such an offshore plant, the core would be expected to melt through the reactor containment (including many feet of concrete) and fall to the ocean floor. From here, one may expect dispersal of some portion of the radioactive inventory as well as biological concentration of a number of nonvolatile fission products, including plutonium.

The principal motivations for offshore barge siting are increased site availability and improved economics rather than major safety improvements. The potential opportunity for remote siting is perhaps the most attractive safety benefit of the concept. Current floating nuclear power plant designs call for water depths of not more than about 70 feet. Thus, in contrast to the Eastern seaboard, the narrow continental shelf along the California coast severely limits the availability of acceptable remote offshore sites. Accordingly, offshore siting is not judged to offer significant potential safety improvements in California's geophysical setting.

5.3.2 LWR System Modifications

Modifications to LWR systems to control core meltdowns must be designed to cope with all of the various phenomena that may lead to

containment violation. The kinds of plant modifications that have been proposed in the literature to control such phenomena typically center on specific subsystems in order to control specified phenomena. Examples of such subsystems are:

- a. Drains to remove water from the path of the molten core to prevent steam explosions.
- b. Core catchers to prevent meltthrough of the containment.
- c. Blowout chambers to alleviate overpressurization.
- d. Filtered venting to remove volatile fission products while relieving pressure by releasing largely purified gases to the atmosphere.
- e. Massive heat-sinks to absorb fission product decay heat until the decay heat generation rate is reduced to the point that conduction and convection processes can remove all subsequent heat from the containment.

These approaches have been included in the systems selected below to demonstrate the plausibility of complete control of meltdown accidents.

Another approach would be to design a simpler, more reliable ECCS capable of reducing core melt probabilities well below existing levels (e.g., as proposed in Ref. 5-2). A melt prevention concept of this type has been included among the system concepts evaluated.

5.3.3 Subsystem/Systems Approach

It was found early in the literature search that most published concepts address but one of the four phenomenological design problem areas, listed in Section 5.2. Few authors addressed complete control of core-melt accidents. Safety modifications intended to achieve less-than-complete control of meltdown are termed subsystem concepts. In the evaluation approach of this study, combined or modified concepts were synthesized from the literature to achieve control of the four key phenomena.

Table 5-2 lists some of the means proposed for solving the major safety problems associated with a Class-9 accident. Some methods contribute to the solution of more than one design problem. For example, employing a core catcher to avoid core penetration of the containment also prevents steam and CO_2 production due to core-concrete interactions. Venting gases to reduce containment pressure also removes heat associated with the decay of fission products in the vented gases.

The "miscellaneous" solutions for steam explosions include inhibitors aimed at preventing explosions by addition of surfactants to any cooling water that might be subject to core intrusion. Though the physical processes associated with steam explosions are not well understood, reducing the surface tension of the water appears to diminish the probability of such events. Heating the water to near its saturation temperature also appears to reduce the probability of steam explosions. Core guides, designed to channel the molten core to dry areas, could also double as core catchers, while satisfying their basic function of preventing steam explosions.

Massive melting bed core catchers have been proposed which would form a low melting point eutectic with the molten UO_2 fuel and provide enough heat capacity and latent heat of fusion to reduce the probability of heating the containment floor above its failure temperature.

Table 5-2. Published Methods on Core Melt Safety

DESIGN PROBLEM	SOLUTION	METHOD
STEAM EXPLOSIONS	BLAST SHIELDS DRY SUBREACTOR SPACE MISCELLANEOUS (BMI-1910)	SHIELDS AROUND BLAST LOCATIONS DRAINS BENEATH PRESSURE VESSEL INHIBITORS; SATURATED WATER; CORE GUIDES
CORE PENETRATION OF CONTAINMENT	CORE CATCHER WITH HEAT REMOVAL	WATER-IMMERSED CORE-CATCHER ACTIVELY-COOLED CORE CATCHER MASSIVE MELTING BED
VAPOR-GENERATED OVER-PRESSURIZATION	COOL DRY CORE CATCHER BLOWOUT CHAMBERS VENTING SEPARATE WATER FROM ALL DECAY HEAT AVOID CONCRETE-CORE INTERACTIONS AVOID CONCRETE HEATING BY VOLATILE F.P. H ₂ REMOVAL	ACTIVELY-COOLED CORE CATCHER TUNNEL OR CHAMBER FILTERED VENTING THERMALLY DISTANT WATER DRAIN VOLUME CORE CATCHER VENT HOT GASES, HEAT REMOVAL BASALTIC CONCRETE SPARK PLUGS/O ₂ SOURCE CHEMICAL REACTIONS
LONG-TIME HEAT REMOVAL	REMOVE HOT GASES FROM CONTAINMENT PASSIVE COOLING OF CONTAINMENT SURFACE ACTIVE COOLING	VENT HOT GASES TO HEAT SINK/VENT; WET MELT IMMERSED STEEL CONTAINMENT BUILDING; WET MELT POROUS CONCRETE HEAT SINK MANY

Blowout tunnels or chambers for pressure relief may be a practical way of preventing vapor-generated overpressurization, since the total vapor generated during a meltdown is not extreme, although at nominal pressures, it may be several times the volume of conventional PWR containments.

Filtered venting concepts utilize a system of filters that are designed to absorb (or trap) fission products from the containment atmosphere and release the cleaned gases to the atmosphere. Current safety systems often include circulation of containment gases through filters, but active systems of this type have not been explicitly considered here because of the desire for a passive system. It would be very difficult to trap the radioactive noble gas components of the fission products in a passive filter, and it has been assumed that the noble gases would be released. The biological dose conversion factors are relatively low for human exposure to radioactive noble gases. Consequently, if venting was restricted to noble gases alone, the dose delivered to the population surrounding an accident site, and the fallout and permanent poisoning of the landscape would essentially be eliminated for all practical purposes.

Basaltic concrete refers to a concrete with aggregate composed mainly of simple oxides such as CaO , SiO_2 , and Al_2O_3 . Use of this type of concrete in foundation slabs for containment structures would minimize the production of CO_2 and substantially reduce the potential volume of noncondensable gases available for release to the containment atmosphere.

Ignitors such as spark plugs could be used to produce hydrogen combustion, or localized small explosions, before the buildup of dangerous concentrations could occur. The available air, however, may only be sufficient to burn about 10 percent of the hydrogen produced, according to one estimate. The deliberate introduction of added oxygen to complete the hydrogen removal would permit a more thorough elimination of hydrogen; however, a mechanism would have to be available

to condense the resulting steam and to eliminate or absorb the heat of combustion. Chemical reactions could also be used to remove hydrogen from the containment atmosphere, assuming the application of similar provisos for removal of energy associated with the chemical reactions.

For the problem of long-time heat generation, venting of hot gases from the containment is a possible approach. A wet melt concept has been proposed whereby a submerged core catcher would produce a continuing supply of steam to mix with the atmosphere and its fission products. This would provide a restraining mechanism for temperatures that are developed in the containment atmosphere.

5.4 SYSTEM CONCEPTS

In this section, synthesized concepts from the nuclear safety literature have been presented that appear to have relatively high potential for controlling meltdown accidents in LWRs. Modifications are suggested to yield conceptual systems that are designed to be capable of controlling all phenomena leading to containment violation. The resulting combinations of published concepts are not unique. They appear to be technically feasible, but there are engineering difficulties that require investigation associated with each of the concepts presented. Questions of cost, effectiveness, licensing impacts, and commercialization schedules were beyond the scope of this study and have not been evaluated.

5.4.1 Dry Melt Plus Massive Melting Bed Concept

The key elements of this concept (Ref 5-3) are depicted in Figure 5-1. The concept features a design to keep water away from the dry well of the containment structures, i.e., a dry melt to reduce steam production, and a large melting bed core-catcher. As indicated, drains keep the space under the reactor dry, reducing the potential for steam explosions that might be expected upon reactor pressure vessel

Figure 5-1. Dry Melt Plus Massive Melting Bed

meltdown. The molten core, with its inventory of nonvolatile fission products and associated decay heat, would be caught in a melting bed of oxide material that was free of water and carbonates. The core and bed material have been designed to form a eutectic with a melting temperature low enough to avoid damaging the containment walls. The bed must have sufficient thermal capacity, including its heat of fusion, to assure a permanently stable configuration for the molten materials in order to prevent meltdown of the containment. Note that the design depth of the melting bed is comparable to the total distance to which cores are expected to penetrate the earth in uncontrolled meltdowns, primarily for the purpose of assuring that the downward migration of the core is stabilized within the melting bed before it results in violation of containment integrity.

Close examination of this concept reveals that long-term heating from the melting bed may eventually induce boiling of the water in the sump ditch, leading to containment overpressurization. In addition, there is no provision for long-term cooling of the containment atmosphere. This is important, since a sizeable fraction of all decay heat comes from volatilized fission products in a core-melt accident. Without provision for long-term cooling of the containment atmosphere, extreme temperatures could be reached within the containment with consequent loss of integrity. (Table 5-3 summarizes the strong and weak features of this concept.)

Table 5-3. Dry Melt Plus Massive Melting Bed Concept

Key Design Problems and Proposed Remedies

Steam explosions

Blast shields

Drains beneath pressure vessel

Core penetration of containment

Massive melting bed

Vapor-generated overpressurization

Dry melt

Carbonate-free melting bed

Long-time heat generation

Active cooling

Unresolved Problems

Late-time steam production due to heating of water in sump ditch

Lack of cooling for containment atmosphere

In order to address all four of the key design problems of core-melt accidents, additions to the concept described in the previous

figure have been incorporated in Figure 5-2. To prevent boiling of the

Figure 5-2. Dry Melt Plus Massive Melting Bed (Modified)

cooling water collected in the sump ditch, the water could be drained to a thermally remote location. If the water were part of the fluid inventory of the emergency core cooling system and/or the containment cooling system, the draining operation could be initiated by the operator when loss of control had clearly taken place, (e. g., failure of the ECCS) in the hypothesized accident leading to meltdown. Alternatively the space beneath the reactor could be maintained permanently dry, and the distant sump could be designed to be an integral part of the ECCS and containment cooling system.

Additional expansion volume or filtered venting could be optionally included as a possible last resort against containment over-

pressurization in the event that excessive vapor generation could not be avoided. Hydrogen production, in particular, might prove to be a problem, since much of the Zircaloy of the reactor fuel rods would be expected to react with steam prior to meltdown.

A suggested solution for long-time heat generation effects on the containment structure is the use of a huge heat sink within the containment, possibly incorporated as part of the containment walls. Thick porous concrete inner walls could provide such a heat sink for gaseous and volatile fission product decay energy. Large-aggregate ("popcorn") concrete has previously been considered for reactor safety applications (Refs. 5-4, 5-5). Its very high permeability provides for distribution of decay heat to the entire thickness of porous concrete. To illustrate the quantity of concrete required, 25,000 tons of popcorn concrete would have sufficient heat capacity to absorb all the decay heat of both volatile and nonvolatile fission products for 10 days. Although this amount of concrete is more than that currently used in the walls of power reactor containments of current designs, it does not appear to be prohibitive from a design standpoint. In a properly engineered system, more extensive thermal studies would have to be performed to size the heat sink. To facilitate decontamination in the event of minor incidents, a smooth steel inner surface might be used for the inner surface of the containment structure, with numerous, pressure-actuated ports included for admitting vapors into the concrete in the event of a severe accident.

Technical uncertainties relevant to this concept include: the melting bed technology, which has yet to be developed; damage to walls by β -ray and γ -ray activity from condensed volatile fission products; and nonuniform heat dispersal within containment (i.e., local hot spot development).

5.4.2 Wet Melt Plus Steel Caisson Concept

Reference 5-6 suggests that a thin steel containment shell, immersed in a pool of water acting as a heat sink, may be adequate to

carry away all required decay heat, providing that steam condensation serves to enhance heat transfer on the inner surface. This concept is illustrated in Figure 5-3. The thickness of the steel shell would be in the range 1.5 to 15 in. to be thermally acceptable. The concept calls

Figure 5-3. Wet Melt and Steel Caisson

for a "wet melt" (e.g., water-immersed core catcher) to provide a reliable supply of steam so that the air in the containment is not overheated by the volatile fission products. Of course, the water-immersed core catcher would also prevent core penetration of the containment floor.

Table 5-4 summarizes the approach of Ref. 5-6 to the design problem areas. This concept in itself may not provide complete protection against overpressurization, primarily because of the hydrogen gas problem.

Table 5-4. Wet Melt Plus Steel Caisson Concept

Key Design Problems and Proposed Remedies

Steam explosions;
blast shields, etc.

Core penetration of containment;
water-immersed core catcher

Late vapor generation;
steam condensation on containment wall

Long-time heat removal;
immersed steel containment building, wet melt

Unresolved Problems

Containment overpressurization by hydrogen gas

Water-immersed core-catcher design

If steam condensation alone is found to be incapable of preventing excessive containment pressures, auxiliary means could be provided to reduce pressures. These could include hydrogen removal, blowout volumes, or filtered venting to the atmosphere.

A number of areas of technical risk are associated with this concept. For example, the concept may be susceptible to steam explosions as a result of the proposed water-immersed core catcher. Water-immersion as a means of preventing core catcher meltthrough is not a proven concept, and many complex phenomena are involved. The effectiveness of the condensation concept for vapor control should be evaluated with detailed calculations. The potential problem of localized fission product hot spots should also be analyzed in detail.

5.4.3 Pressure Management by Expansion Chambers and Filtered Venting

The use of supplemental expansion chambers and/or filtered venting to accommodate vapor generation in an underground siting concept (see Figure 5-4) is discussed in Ref. 5-7. The basic ideas,

Figure 5-4. Expansion Chambers and Filtered Venting Concept

however, could be used to support the containment of core-melt accidents in aboveground LWRs as well as in underground facilities. Through use of these methods, the problems of vapor-generated overpressurization would be attacked by allowing gases to be released from the containment to one or more buffer volumes through pressure suppression and relief tunnels. If pressures were to subsequently increase beyond acceptable limits, the system would be designed to vent through a high stack, with filtration provided for the volatilized fission products. It is conceded that a filter capable of collecting noble gases would probably not be practical for a passive system, since cryogenic cooling would be required in order to trap the noble gases.

Though the concept reduces the risk of overpressurization by means of a controlled and filtered venting mechanism, it does increase risks to the public in the sense of allowing the venting of some fission products rather than providing complete containment. However, the biological conversion factors are very low for exposure to radioactive noble gases; hence, the consequences for their release would not be expected to be substantial.

For example, interpolating results from the APS review of LWR safety (Ref. 5-1), under severe core-meltdown accident conditions, less than 10 deaths from cancer might be expected to occur as a result of release of all the noble gas fission products from the reactor; and these would take place over a period of up to 30 years. By comparison, the total cancer mortality estimate made by the APS for this accident was of the order of 15,000 deaths. Thus, restricting radioactivity releases from severe accidents to essentially only those associated with the radioisotopes of the noble gases could substantially reduce the overall impacts of such accidents.

5.4.4 Wet Melt Plus Expansion Chamber/Filtered Venting Concept

Neither steam explosions nor the problem of penetration of the containment floor by the molten core are discussed in Ref. 5-7. This may have been due to their reliance on underground siting features. An aboveground version, however, must address these problems if equivalent containment is to be achieved. Figure 5-5 depicts such a version, wherein the threat of steam explosions is met by methods described earlier, and the problem of containment meltthrough has been addressed by means of the water-immersed core catcher of Ref. 5-6. In this modification, returning condensate from the expansion volume would aid in keeping the core catcher submerged for long-time cooling.

Table 5-5 summarizes features of the concepts introduced in Figure 5-5 and highlights the potential problems for above ground applications.

Table 5-5. Wet Melt Plus Blowout Concept

Proposed Remedies

Steam explosions
Blast shields, etc.

Core penetration of containment
Water-immersed core-catcher

Vapor generated overpressurization

Expansion volume
Filtered venting

Long-time heat generation
Expansion volume
Filtered venting

Potential Problems

Steam explosions

Water-immersed core-catcher design

Figure 5-5. Expansion Chambers and Filtered Venting Concept (Modified)

Again there are several areas of technical uncertainty that deserve detailed engineering calculations. They are steam explosion phenomenology, design of immersible core catchers, heat transfer and dispersal from the containment, and containment wall damage from plate out of volatile fission products.

5.4.5 No Melt: "Passive Containment System"

A different type of concept - averting core melts through passive (though complex) means by the use of a specially designed PWR containment system (see Figure 5-6) - is proposed in Ref. 5-2. In this patented scheme, automatic flooding of the core and spaces around the reactor vessel has been designed to occur by use of water from high pressure tanks in quantities stated to be sufficient to overwhelm any accident. Although a fairly complex configuration of valves, tanks, etc., is employed, this system is purportedly simpler than a conventional ECCS and is characterized as passive in that it requires no active pumping system. It should be noted that this system employs conventional pumps, sprays, etc., to cope with accidents less than a LOCA.

The intended operation of the Passive Containment System may be briefly described as follows. In case of a LOCA, the primary coolant blows down into the shaded region of Figure 5-6. When the pressure exceeds a pre-set level, in principle, specially designed diaphragms rupture, admitting blowdown steam into the deluge tanks and forcing steam through the deluge tank water. Upon equalization of pressure, the deluge tank water is designed to flow into the shaded region, thereby completely flooding the primary containment. Meanwhile, when appropriate pressure differentials trip check valves, high-pressure water from other sources (e.g., refill tanks) would be injected into the hot and cold legs, filling the reactor. It is believed by the authors of Ref. 5-2 that, when the reactor and primary containment are full, the pressure vessel will remain filled passively. The remaining problem is long-term heat removal from the flooding water. This is accomplished by means of passive, gravity-convection driven heat exchangers to an outdoor pond.

No Melt: "Passive Containment System"

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Figure 5-6. No Melt: "Passive Containment System"

This system has been taken beyond the conceptual stage, but has not yet been subjected to the intense scrutiny applied to conventional ECCS concepts in the course of licensing procedures. However, it is representative of a general class of potentially very important safety augmentation concepts. These alternate ECCS schemes would reduce the risk of meltdown accidents by improving the reliability of the safety systems and thereby attempting to eliminate meltdowns, rather than minimizing the consequences after an occurrence. The concept is claimed to be less expensive than a conventional containment and ECCS.

A potential problem is associated with the proposed passive heat exchangers for decay heat removal. With the small heat exchanger volume available within the containment, adequate heat exchange capacity may be difficult to provide passively. This should be one of the key problems to be investigated in further studies of this concept. If passive heat exchange to the outdoor cooling water proves to be inadequate for the Passive Containment System, an active system (e.g., circulation pumps) could be added to the external heat exchanger.

5.5 COMPARATIVE CONCEPT EVALUATIONS

5.5.1 Evaluation Criteria

Seventeen evaluation criteria were developed for qualitative ranking of system concepts, as follows:

Simplicity

Simple moving parts

Passivity

No operator action

No electrical/mechanical support

Confidence

Analysis insensitive

Experiment insensitive

Feasible to demonstrate

Improvement against external events

Acceptable to public

Licensability

Maintains containment

Retains ECCS

Safety not jeopardized

Regulatory acceptability

Cost

Reasonable test cost

Reasonable Implementation cost

Cost of Accident

Saves reactor

No balance-of-plant damage

No long-term maintenance

The criterion "Improvement against external events" refers to a design objective for the safety system of providing significant benefits, even under extreme conditions such as war or sabotage.

The criterion "Safety not jeopardized" refers to avoidance of advanced system design features that could cause new safety problems or reduce the effectiveness of other safety system components, such as ECCS elements.

"Regulatory acceptability" refers to the possibility of a system concept violating the extensive, codified safety regulations of the NRC (e.g., 10 CFR50). These regulations call for such safety features as emergency cooling water, an emergency core cooling system, and an essentially gas-tight containment. A safety system providing for a "dry melt" (i.e., draining of spaces beneath the reactor), might be in violation of these regulations. Controlled venting may also be considered in violation of current containment requirements, even though it would only be employed to prevent more severe releases as a backup feature to the conventional engineered safety features of the plant.

As a desirable design goal, the implementation cost of a meltdown control system should represent a small fraction of conventional LWR total plant cost, e.g., from 1 percent to, perhaps at most, 10 percent. The terminology of "reasonable" costs has been used to imply that range. This definition of "reasonable" was rather arbitrarily made and subjectively

used for this review rather than founded on well-defined cost-effectiveness considerations.

Clearly the 17 criteria are not all of equal rank. The first three categories - simplicity, passivity, and confidence - correspond to technical goals and are considered to contain the most important of the listed criteria.

An example of inequality of criteria is in the cost categories. "Cost" refers to costs that must be borne to implement the concept. "Cost of accident" refers to hypothetical costs that would be incurred only in the event of a major meltdown. The latter costs should be weighted by the (low) probability of a major accident; thus, their importance could be weighted quite differently by different people.

5.5.2 Application of Evaluation Criteria

An attempt has been made to assign relative ratings to the four system concepts discussed above for each of the 17 evaluation criteria. The results are shown in Table 5-6. The rankings shown were derived on the basis of a subjective evaluation, rather than a quantitative determination. The results provide a qualitative overview of the relative merits of each of the systems analyzed, for each evaluation criterion.

As noted earlier, the criteria vary widely in importance. Without the benefits of supplemental studies to define the costs, effectiveness, regulatory and licensing impacts, and commercialization schedules for each system concept, it was deemed imprudent to specify aggregate rankings for the four systems.

Table 5-6. Application of Evaluation Criteria

Criteria	Melting Bed	Wet Melt + Steel Caisson	Wet Melt + Blowout	Passive Containment System
No Operator Action	E	E	E	G
No Electrical/Mechanical Support	E	E	E	F
Simple Moving Parts	G	G	G	G
Analysis Insensitive	G	F	G	G
Experiment Insensitive	G	F	G	G
Reasonable Test Cost	F	F	F	G
Reasonable Implementation Costs	F	F	F	E
Feasible to Demonstrate	F	F	F	G
Maintains Containment	E	G	G	E
No Balance-of-Plant Damage	E	E	E	E
Saves Reactor	P	P	P	E
Retains ECCS	E	G	E	P
No Long-Term Maintenance	F	F	F	F
Acceptable to Public	E	G	G	F
Safety Not Jeopardized	E	F	G	E
Regulatory Acceptability	E	G	F	E
Improvement Against External Events	E	G	G	P

E - Excellent

G - Good

F - Fair

P - Poor

A literature survey was performed to identify proposals for improving the safety of above ground power reactors by containing or mitigating the consequences of a Class-9, or core-meltdown, accident. The study, restricted to the consideration of light-water power reactors, sought alternatives which could compete favorably with the costs and benefits offered by underground siting.

Four key problems were identified which could lead to containment failure. These were classified as steam explosions, overpressurization, core melthrough, and long-term heat generation. The concepts offered in the literature to overcome these problems include blast shields, core catchers, blow-out chambers, controlled venting, and passive heat sinks. Unfortunately, most of the specifically proposed solutions were found to be incomplete in that not all of the principal design problems associated with control of a core-meltdown accident were addressed. Therefore, this portion of the study undertook to synthesize selected combinations of these concepts which might more completely contain a Class-9 accident.

There are technical uncertainties associated with each proposed concept. These have been noted, and the nature of the research required to resolve them has been identified. When technical feasibility has been established, the problems of resolution of supplemental questions of cost, safety, regulatory and licensing impacts, and schedule for commercial power plant application will then require more thorough system analyses and engineering development.

From consideration of the system solutions identified above, it was concluded that complete control (or reliable prevention) of core-meltdowns is within the realms of technological feasibility, despite technical uncertainties that do exist.

Among factors favorable to the prospects for achieving control are the following:

- a. Core-melt sequences and many of the basic phenomena involved have been studied in detail, and there is significant literature on the subject.
- b. The major phenomena that can lead to containment violation - heat production and gas production - are inherently simple in nature.
- c. It appears that with sufficient thermal mass to accommodate heat produced, and sufficient volume to accommodate gases produced (with appropriate engineering design), meltdowns could be controlled through conventional applied engineering design methods. Whether this would be cost-effective in terms of public safety is a larger question requiring further study.
- d. Simple, passive means of preventing meltdowns have not been given adequate attention in the past but appear to be promising in terms of cost as well as efficacy.

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6.0 PROGRAM DEVELOPMENT FOR DETAILED STUDY PHASE

Based upon the results of the Initial Program Definition Studies conducted by Aerospace, plans were developed by Aerospace/ERCDC for more definitive investigations of underground facility feasibility, cost, and effectiveness in the Detailed Study Phase of this program. The prime objective of these studies will be the derivation of credible numbers for insertion into a decision analysis framework. To this end, architect-engineers and reactor safety experts will be employed to address the important design issues discussed in the earlier sections of this report.

Realism will be emphasized in these studies. Specific engineering designs for rock-mined and berm-contained reactor power plants will be prepared. Designs will be prepared based upon geological siting conditions which have been selected on the basis of their realistic availability within the state. Problems of construction, operation, and maintenance will also be addressed in relationship to issues of cost, convenience, operational and public safety, etc. Analyses of public consequences resulting from severe reactor accidents will be based upon population characteristics and meteorology which are also realistic for conditions within the state. The California Division of Mines and Geology will participate in the characterization of geological regions within the state which might be suitable for underground siting of nuclear power plants in accordance with geotechnical and engineering criteria for evaluation of siting concepts.

Cost of construction for aboveground and underground plants (both rock and berm-contained) will be evaluated. Early evidence suggests that the precision of the engineering cost estimates based upon conceptual designs will be reasonably good (perhaps within 15 percent) if the assumptions of the architect-engineers are valid. If, on the other hand, delays are introduced into the construction schedule (e. g., perhaps due to regulatory, intervention or site-related problems), significant variations from the cost estimates might occur. The magnitude of these variations and the probability of their

occurrence will also be estimated. Operation and maintenance requirements will be considered in the first cost evaluation. Cost of transmission lines and distribution facilities will not be considered, since they would be common to aboveground and underground plants sited in the same locale. No specific attempt to calculate costs of power to the consumer will be made because of the many elements of the rate base which will not be addressed. However, it may be possible to derive busbar energy costs (for approximate relative costs to the consumer) based upon estimates of relative total plant costs and availability factors.

From underground plant accident sequences developed in the second-half ERCDC studies, radioactive release rates will be calculated and pathways to the biosphere will be identified. Consequences to the public health in terms of deaths (prompt and delayed) and illness will be calculated. Economic loss factors such as unavailability of arable lands and the necessity for evacuation and decontamination procedures will be estimated.

Environmental effects in the design, construction, and operational phases of the plants will be compared for surface and underground nuclear facilities, as well as effects associated with postulated accidents. Each effect will be separately identified. However, only differential effects (i. e., those which are different for aboveground and underground plants) will be dealt with in detail. In each case, environmental effects will be identified as site-dependent or site-independent.

6.1 SUMMARY OF APPROACH FOR DETAILED STUDIES

In performance of the technical definition of the underground concepts for the program, baseline designs for underground facilities will be prepared and cost of construction estimated by two contractors. One will produce a berm-contained design and the other a mined-rock design. The design goal for these underground facilities is to minimize the consequences of radioactive fission products which might be released to the public from a reactor core melt-down (Class-9) accident. However, as an additional design objective, the underground plant designs should not increase public risks for Class-8 Design Basis Accidents (DBA) above those for a surface plant. The

basic motivation for conducting two studies is to investigate the principal implications of different facility designs for increased public safety. An iterative review and evaluation will be conducted (between the contractors, the ERCDC project team, and the Aerospace System managers) of the preliminary design specifications described in Section 6.2. The configurations to be defined will, at minimum, reflect the functional requirements of various plant systems, operation and maintenance, and accident consequence control. The latter requirements imply a clear recognition of the concept of "containment," embodying both structures and a functional level of constraint against fission-product releases.

The compatibility of the general plant configurations with the Nuclear Regulatory Commission's existing requirements and criteria will also be assessed. A design objective of the underground facility design is to achieve licensability for the underground plant against NRC requirements for Class-8, or Design Basis Accidents (DBA) with no increase in public risk, when compared to surface nuclear power plants.

6.1.1 Accident Management System Design Level Definitions

The basic design philosophy provided for the direction of the architect/engineers reflects the judgment that total containment of all fission products from a Class-9 reactor accident is not a reasonable technological objective for an underground containment structure without provision for energy removal systems, which probably then would be outside of reasonable cost objectives. Consequently, in order to clearly define facility design and public consequence analysis requirements, four levels of underground nuclear power plant facility design have been defined, as shown in Table 6-1. These incorporate design measures with successively increasing requirements for conservatism (with concomitantly increased cost and complexity) in managing and/or controlling and containing a Class-9 core-meltdown accident. These categories have been defined for the Architect/Engineer (A/E) task descriptions in order to specify the levels of accident management required for design tasks. The baseline for the A/E conceptual designs is the Design Level 2 Accident Management System, with the Design Level 3 concept specified for a sensitivity study.

Table 6-1. Levels of Underground Plant Facility Design

- Level 0 - Incorporation of only conventional Class-8 surface facility design features and methodology in an underground environment.
- Level 1 - Conventional surface facility Class-8 design with protective isolation devices on all tunnels, shafts and other penetrations to the surface.
- Level 2 - Elementary Class-9 accident mitigation system using a pressure relief mechanism (high-permeability tunnels, expansion volumes, etc.) with a directed path to the earth medium surrounding the facility, in order to achieve pressure and fission product management through utilization of the natural external environment of the underground facility.
- Level 3 - Class-9 accident mitigation systems with an added pressure management system incorporating a controlled, filtered venting system.
- Level 4 - Class-9 accident control with complete fission product retention within underground containment using core catchers and long-term energy dissipating systems, as necessary.

Design Level 0 was defined in order to illustrate the case of a minimal perturbation to a surface plant design. Though more protection is provided than for the Design Level 0 concept, simply putting closures on tunnels, as per the Design Level 1 concept, may not provide sufficient isolation of released fission products. Design Level 1 may also fall short of desirable levels of fission product containment for public safety, because of expected pressure conditions, wherein the leakage that might ensue in the event of containment failure could be uncontrolled and indeterminate as to location and magnitude of release.

Design Level 2 was prescribed as the baseline concept for this study program, in that it was expected to provide reasonable engineering assurance that the path of leakage could be engineered so that natural filtering through the earth would occur in a manner amenable to analysis. The Design Level 3 concept goes one step further and incorporates added engineered devices, in order to provide an engineered path of known permeability, porosity, and filtration characteristics from the containment to ensure continuous fission product and energy transfer out of the containment structure.

The Design Level 4 concept was considered to require technological skills beyond the scope of this conceptual study and possibly beyond the present architectural and engineering design technology for today's nuclear power plants. The concept envisions such advanced concepts as core catchers and active heat removal mechanisms from the containment structure to the external environment. The definition of the Design Level 4 Accident Management System was included in contractor work statements in order to provide completeness for the definitions, as opposed to presenting a contractual performance requirement.

6.1.2 Cost Uncertainty Factors

In connection with the development of plant layouts for baseline and alternative underground facility concepts, major cost factors unique to underground siting will be identified. Several elements of these cost estimates are recognized to be uncertain for the level of design associated with this study. For example, underground excavation costs are highly dependent upon conditions of site geological materials. As a result, excavation costs often exceed preliminary estimates. Only generic site descriptions will be available for these studies, thereby increasing cost uncertainties in this area. Certain design objectives may require novel technical solutions. For example, large, low-leakage, access tunnel closures may be included in this category.

Obviously, it is not possible to estimate the cost of first-of-a-kind "pioneer" systems as precisely as estimates for frequently replicated systems. It is an objective of these studies to identify (a) the inherent uncertainties

in the major cost areas and (b) to mobilize techniques, such as probabilistic cost estimating, to cope with the uncertain factors of such new system concepts.

6.1.3 System Definitions for Underground Facilities

Detailed definition of those systems or concepts that differ markedly from conventional surface facilities will be provided. For example, if the main turbine condenser is included in the underground portion of the plant, at an elevation considerably below the cooling water source (as might be the case with some alternative containment concepts), either pressure relief mechanisms for condenser coolant or high-pressure condensers must be adopted for the design. Important changes required in system components, when compared to surface facilities, will be identified for the baseline and alternative concepts.

Given the fundamental public safety objective of these studies, the surrounding soil or rock, and some other systems unique to underground construction, such as access tunnel closure valves, etc., may become safety-related elements of the structures in the event of Class-8 Design Basis Accidents, in addition to the protective role which they would play for Class-9 accident control. This may require a redefinition, or expansion, of the Engineered Safety System elements currently identified by the NRC for surface nuclear facilities. These items will be identified, and the NRC regulatory implications will be noted relative to Class-8 accident designs during the course of the facility design studies.

6.1.4 Conceptual Designs

Based on the results of the foregoing efforts, conceptual plant designs will be developed for the baseline power plant concepts. The level of design detail provided in the studies will be commensurate with the broad, overall objectives of the ERCDC studies. These require assessments of surface and underground nuclear power facilities with respect to implicit public safety capabilities, etc., within the funding and timing limitations for these studies. The level of detail required for various plant systems depends on at least five factors:

- a. The definition required for subsequent safety analyses
- b. The sensitivity of the system to cost variation
- c. The sensitivity of the system to site conditions
- d. The extent to which the component or system has been altered from surface configurations
- e. The impact on construction scheduling.

It will be the responsibility of the selected architect/engineer organizations (or teams) to provide technically realistic plant definitions and cost evaluations that reflect their experience and engineering judgment. It is recognized that cost estimates based upon conceptual designs are indeed approximations, and, in studies such as these, will depend heavily on experience and judgment. Thus, the level of final design for these studies should be consistent with the need for, or ability to, provide accurate cost estimates. In accordance with this objective, additional design data will be developed as required to support the sensitivity analyses of alternative design concepts.

6.1.5 Scheduling and Costing

During construction of any underground facility, personnel and equipment access to working areas will be restricted, as compared to conditions for surface plant construction. These restrictions are expected to affect construction scheduling. There are other construction steps required in the underground concepts that are not present to the same degree in surface construction. An obvious example is the requirement for major excavations for underground containment and principal auxiliary buildings. This will be a critical path item in construction schedules, and unanticipated problems may arise. Completion of excavation will pace the time span for construction of the entire underground facility.

The most significant aspects of underground construction techniques are to be identified in the course of the second-half of the ERCDC program and their impact on construction schedules assessed. This information will also be incorporated into the preparation of realistic estimates for cash flows for the facility concepts.

Plant design and schedule data will be utilized in estimating major system costs for reference surface plants, baseline concepts, and required sensitivity analyses. Where possible, cost estimates will be based on existing information from construction experience for surface facilities. For those systems that are only slightly modified (or are unmodified) from surface plants, potential costs may be derivable from estimates of the differential requirements of underground construction. Cost estimates will also reflect the information generated from the efforts to define the principal factors contributing to cost uncertainties.

Cost sensitivity analyses will be conducted with self-consistent assumptions in accordance with cost estimation methods approved by ERCDC/Aerospace. Estimated costs for the facility concept variations considered are expected to be developed within reasonable ranges (i. e., $\pm$ one standard deviation) of the most probable costs, where meaningful statistical cost data is accessible for the studies.

6.1.6 Public Consequence Analyses

The other major endeavors to be conducted during the Detailed Phase of the study are (a) the analysis and evaluation of accident and post-accident processes (for postulated severe accidents involving reactor core meltdown) and (b) the determination of relative public consequences associated with the accidents for selected underground and berm-contained nuclear plant designs. This study will include (a) characterization of fission product source terms (within the containment structure), (b) analyses of fission product release paths (transport) for selected site conditions, and (c) analysis of fission product removal mechanisms for selected site conditions. Also to be established are the post-accident environmental/public health effects relative to surface facilities, and any other related long-term post-accident consequences to the public.

A set of accident sequences for underground nuclear power plant designs will be selected, analyzed, and defined for direction of plant design efforts as well as analyses of potential significant public consequences. This effort will include the following:

- a. Identification and assessment of the applicability of previously identified surface plant Class-9 accident sequences to underground facility designs.
- b. Identification of unique accident sequences for the Design Level 2 PWR berm containment and rock facility design concepts
- c. Identification of accident sequence differences (if any) between the Design Level 2 underground PWR facility, the alternative Design Level 3 PWR, and the Design Level 2 BWR concept.

Energy-temperature-pressure time histories, failure modes, time sequences to failure and fission product release and distribution within the facility structures will be evaluated. These evaluations will include:

- a. Definition of the radionuclide inventory of the core at time of accident for the conservative case corresponding to decay energy defined in accordance with American Nuclear Society Standard-ANS 5.1 + 20 percent, for both baseline PWR and alternative BWR cases
- b. Determination of critical meltdown time sequences
- c. Determination of energy release and heat transfer modes; and the time histories and distribution of energy, pressure, and temperature within the facility structures
- d. Determination of time-dependent gaseous and volatile radionuclide distribution and conditions within the facility structures
- e. Characterization of the time-dependent fission product inventory of the molten core, together with related estimates of core energy.

Methods for analyzing the gas dynamics and fluid dynamics of the flow of gaseous and volatile products of the core meltdown process through soil and rock will be identified. The models defined will permit consideration of cases of ideal flow through permeable media; and flow through idealized "channels" in soil, fissures in rock, or leakage paths along construction interfaces with surrounding earth media for facility penetrations (tunnels, shafts, etc.) which interdict the ground surface.

On the basis of the methods identified for analyzing the gas dynamics and thermodynamics of flow through soil and rock, estimates will be provided of (a) the short-term (up to 6 to 10 hr) and long-term flows associated with the identified accident sequences, (b) related energy-temperature-pressure time histories within the facilities, and (c) the corresponding time histories of gaseous releases at the ground surface for the Design Level 1, 2, and 3 Accident Management System concepts.

Methods will be identified and specified for analyzing the transport of fission products from within the facility structures to the ground surface for the ranges of thermodynamic conditions and fission product species and concentrations specified in the contractor's Statement of Work (see Appendix I). The analysis methods will be adequate to define the transport modes under either conditions of idealized seepage or more realistic leakage paths through the earth materials.

On the basis of these methods, estimates will be made of the time-dependent fission product releases at the earth surface, and the corresponding subsurface contamination of earth media and ground water associated with the accident sequences.

A report will also be prepared of an analysis of the environmental consequences of underground or berm containment of selected nuclear plant design variations. This study could be utilized by the Commission in its preparation of an Environmental Impact Report (EIR) pursuant to the requirements of the California Environmental Quality Act of 1969 (CEQA). The CEQA requires the Commission to consider the environmental consequences of its actions that could have immediate (or eventual) effects of significance

to the environment. Effort will be concentrated on those impacts which differ from conventional surface nuclear power plant design criteria. Evaluation shall be based on facility designs as developed and specified in Appendix I. Furthermore, the evaluation will be of a generic nature and, to the greatest extent possible, site-independent.

Environmental impacts associated with occurrence of postulated accidents for underground and berm-contained nuclear power plant concepts will also be determined and evaluated.

The impacts of the radioactive releases on public health, safety, land use, flora and fauna will be evaluated and compared to the consequences of comparable surface plant accidents. This analysis and comparison will also be generic in nature and, to the extent possible, site-independent. Specific attention will be given to the estimation of population exposure to released radioactivity and resultant health effects; both early and long-term fatalities and illnesses will be included in the calculations. The evaluation will include economic considerations.

6.2

DEVELOPMENT OF PROGRAM REQUIREMENTS AND SPECIFICATIONS

A substantial portion of The Aerospace Corporation effort in this Preliminary Analysis portion of the ERCDC program was expended in development of specifications and criteria for the studies of plant designs and evaluations of accident consequences for underground power plants described in Section 6.1. The specifications were developed in order to provide direction for the contractors selected to investigate three major areas of ERCDC concern: (a) design of berm-contained underground structures in soil, (b) design of rock-mined cavity facilities, and (c) analysis of public consequences from core-meltdown accidents which had been postulated to occur in either of the bermed or rock-mined concepts. The following sections describe the philosophical basis for selection of the criteria and specifications for the baseline power plant, subsystem locations, site characteristics, and containment requirements.

6.2.1 Specification Selection Philosophy

The baseline design specifications for underground power plants were selected to produce reasonably conservative designs without inducing the A/E contractors to provide either excessively optimistic or pessimistic cost estimates. The specifications also defined a minimum number of sensitivity studies in which key parameters in the baseline specifications were to be varied so that the impact on system design, cost, and effectiveness could be determined. The ranges of parameters specified for sensitivity analyses were selected to bound the limits of reasonably credible optimistic and pessimistic cost estimates. The specifications developed to guide the A/E contractors in the design of mined-rock underground and berm-contained reactor power plants are contained in Appendices I-1 and I-2.

6.2.2 Power Plant Characteristics

The reactor thermal capacities specified in Appendices I-1 and I-2 are based on the current standard reactor nuclear steam supply systems (NSSS) submitted for licensing by the indicated vendors. The 3800 MW reactor thermal capacity corresponds to a net electric capacity of about 1300 MW, a

size judged to be economically attractive by most large utility systems in the 1990 period. The envelope dimensions and component layouts of the NSSS produced by the three U. S. manufacturers of pressurized water reactors are quite similar, as discussed in Section 4.2 of this report. Of these, the Westinghouse NSSS selected for the baseline design appears to represent the most mature design on the basis of similar units planned, under construction, or in operation. As a result of dimensional similarities, the use of the Westinghouse NSSS as the baseline design will not have any obviously serious impacts on applications of the results to the NSSS of other reactor vendors for conceptual underground design purposes. A sensitivity analysis of the cost variations associated with such selections has been specified for study. The General Electric Co. BWR-6, while having similar external plant physical dimensions to PWRs, differs from them in several important aspects of reactor system and containment design. Therefore, examination of the BWR-6 has been required as a specific design alternative with a separate plant layout to be prepared by the contractors.

6.2.3 System/Component Locations

The recommended system/component locations shown in the specifications of Appendix I relative to surface or underground portions of the facility were established after conducting the preliminary identification and examination of configuration issues discussed in Section 4. The decision to place the NSSS underground or beneath the berm containment was dictated by the large fission product inventory within the reactor core and the very high associated radioactive decay energy at reactor shutdown. The decision to place the waste processing and storage in a subsurface location was primarily due to operational requirements and not strongly based upon safety considerations. The recommended location for these systems was selected primarily due to their close interfaces with the NSSS, and also as a result of their inventories of fission products, stored energy, and corresponding potential for fission product dispersal. The long-term fuel storage and handling facility was recommended for a surface location primarily on the basis of space requirements. The spent fuel pit does not appear to represent a long-term hazard resulting from fission

product release of nearly the same magnitude as the NSSS itself, despite the relatively large fission product inventory in the spent fuel pit.

It can be shown that, as a result of the relatively low-energy density of the fuel rods shortly after shutdown, fuel rod meltdown and massive fission product release (in the event of the loss of cooling water from the spent fuel pit) is a technically improbable event after a brief period of initial cool-down of the fuel rods. However, the subsurface location of the NSSS explicitly provides for fuel-handling and storage for at least two-thirds of the core capacity to conservatively meet the needs of normal refueling cool-down requirements. The economics of maintaining two spent fuel-handling system components separately, one at the surface and another in an underground cavity, could be unattractive. Under these circumstances, a single underground spent-fuel handling system would be a logical alternative.

It was presumed that the facility layout and depth of burial requirements for berm-contained plants would permit the turbine and nuclear steam supply system to be in physical proximity, even if the turbine-generator were located aboveground. Consequently, steam and feedwater line lengths were expected to be comparable to surface plants, irrespective of whether a surface or underground location was selected for the turbine-generator. For the mined-rock concepts, on the other hand, the turbine-generator and main transformer were specified for underground emplacement in the baseline case. This is primarily due to the fact that long, high-pressure steam piping runs between underground steam generators and surface installations of the turbine-generator are expected to be a source of severe economic penalties. These penalties would occur not only as a result of the cost impact of the long high-pressure steam pipes, but also as a result of the related thermodynamic impact on plant performance through heat losses in the long pipes leading to surface locations of the turbine-generator. A sensitivity analysis to more quantitatively assess the nature of the costs to both bermed and mined-rock concepts of surface location of the turbine-generator and main transformers has been specified for the Detailed Studies.

Facilities, such as the plant control center, emergency power generators, etc., were specified for surface location for both of the underground concepts. These locations, however, were subject to discretionary revision by the plant designers based upon further consideration of plant design and cost objectives.

6.2.4 Site Characteristics

The baseline site for the berm-containment concept has been selected in soil, whereas the mined-cavity concept is in rock. The intent of these selections for the siting specifications has been to maximize the effectiveness of the earth-covering above the structures to contain escaping fission products while simultaneously minimizing the difficulty and cost of construction for the two concepts. In the bermed concept, the designation of a moist rather than dry soil as the baseline was made to reflect the most general conditions found in nature. A moist soil also minimizes the difficulty and cost of excavation, since such soil conditions are associated with relatively larger permissible angles of repose than those of dry soils.

Generally, sites would be sought where the water table is sufficiently low so that construction would not be unduly complicated, as is the case for the baseline site. However, in areas of Northern California and parts of the Central Valley, water tables can be sufficiently high to interfere with berm-contained siting. Therefore, the specifications call for a sensitivity analysis of this factor.

6.2.4.1 Soil Models

The three soil models of Appendix I-1 were selected to cover a reasonable range of construction conditions that might be required for a cut-and-cover operation. They were meant to be appropriate to the types of soil that might be encountered in either the Central Valley or the desert environments. However, individual physical characteristics for the material properties were selected to present specific engineering problems that were expected to have an impact on construction costs. With the specified variations of the three models, the cost sensitivity of the dominant geologically related variables impacting construction costs are expected to be obtained.

Soil Model 1 was selected to be generically similar to a typical California Central Valley site. The physical properties were selected to represent a case where the upper layers of the soil section were undesirable for use as fill material or for structure foundations. Soil Model 1 was felt to be suitable for founding a moderately deep cut-and-cover facility, however. The rationale for selecting this model for the baseline case was that the site selection for a berm configuration could well be predicated on criteria for providing a graded backfill selected for its fission product transport properties rather than ease of construction. This might impose a requirement for using imported material from a "borrow" location some distance from the site.

Soil Model 2 was developed to represent a more ideal and desirable siting condition. The cut material was selected to be suitable for backfill. A clay layer was included for ultimate use as a relatively impermeable outer shield to inhibit gas leakage from the facility. Bearing capacities of the soil are suitable for foundations at depths below 85 ft.

Soil Model 3 for the berm site represents a condition that is believed to be fairly common in siting areas of interest. This is the case where there is a soil layer of shallow depth overlying a metamorphic rock mass. The model specifies a shale rock mass with steeply dipping beds. The top 40 ft of the shale is highly weathered, while the section below 50 ft is competent. Excavation in the rock can be accomplished by quarrying techniques. The upper 40 ft of rock would require extensive supplementary support - probably in terms of rock bolts - to hold the steeply dipping sides of the excavation in place.

6.2.4.2 Rock Models

The Aerospace Corporation has, in conjunction with its Air Force programs, participated in nationwide large-scale siting programs to determine the practicality of utilizing high-quality, stringent rock model

specifications for siting of advanced strategic weapon systems. These programs culminated in the "Siting Concepts and Investigations" field evaluation program conducted by Ken O'Brien and Associates of Long Beach, California. This program involved exploratory drilling and rock property determinations for a large number of sites across the United States.

The investigated areas included among other sites in the Western U.S., a region of 150 nmi radius around Castle Air Force Base, California, which involved a significant portion of the Central Valley area north of Fresno, California. A review of this portion of the study illustrated that high-quality rock sites were available. Accordingly, Rock Model 1 of Appendix I-2 was selected on the basis of data derived from this siting program. The core log and material properties of the site selected represented more or less median properties for the region described above, i.e., it was neither the best nor the worst site found within the region. Several other sites investigated in the Air Force survey had more suitable rock qualities that could have been viewed by some as being unrealistically high in quality.

Rock Model 1 was selected for the baseline analysis. Rock Models 2 and 3 (also of Appendix I-2) were selected on the basis of the criteria of cost of construction. Rock Model 2 represents a reasonable upper bound of rock properties that may be expected to be representative of an extensive rock mass to be excavated. From a construction standpoint, it represents an excellent site, but one that may be difficult to locate when considering other factors involved in site selection.

Rock Model 3 represents a poor quality rock site. Its selection would probably be based on overriding considerations other than geology. It was selected to represent the lower limit of site acceptability on the basis of construction cost considerations.

6.2.5 Consequence Analysis Parameters

In selecting the specifications and program requirements for the Public Consequence Analysis Program, the same power plant characteristics, plant configurational specifications, and siting data were prescribed as were given to the architect/engineers designing the surface and underground facilities. Some additional demographic and meteorological site definition and specifications of accident conditions were needed for completeness, however.

Site meteorological conditions and demography were specified very generally. The results of the brief study of site characteristics described in Section 3.2 were analyzed. On this basis, it was estimated that the probable regions of interest for siting would divide into about four geographical zones within the state. Two basic geographic regions were considered: the great Central Valley and the Eastern Desert regions bordering the Colorado River. Within the overall length of the Central Valley region, it was evident that several significant meteorological subregions existed. Based on this observation, it was estimated that the Central Valley could be subdivided into three meteorologically similar zones, with one additional zone specified for the Eastern Desert regions.

Further subdivision was specified for each of these zones into rural and remote elements. The meteorological and demographic specifications were purposely left general, until more geophysical detail could be developed in the Detailed Study Phase of the ERCDC program.

Accident conditions were specified in terms of the decay heat rate output for the reactor and its radionuclide inventory. A conservative estimate for the decay heat rate was taken based upon the American Nuclear Society Standard (ANS 5.1) developed for this parameter. Based upon a review of experimental and analytical data for estimation of radioactive decay heat production in a reactor, it was felt that for conservatism the ANS 5.1 Standard should be increased by 20 percent to allow for an uncertainty of about one standard deviation in the estimated decay heat rate.

The development of final specifications of the radionuclide inventory at the time of the accident was charged to the contractor in order to permit him to evaluate all the decay energy analysis requirements as well as the fission-product transport and dose-conversion relationships before making a final selection of the most significant radionuclides to be included in the study. However, specifications were given to the contractor to ensure that the inventory defined should be the most severe that would be expected at any time during the operation of the reactor. Though this might vary for some conditions, the most likely combination was felt to be end-of-life conditions for the spent fuel, with a fully matured fuel load, which would have undergone at least two previous refueling cycles. Under these conditions, the oldest third of a PWR's fuel load would be about three years old, and the newest third about one year old.

6.3 GENERAL OBSERVATIONS

The basic philosophical guidelines which motivated the preparation of contractual documents for the Detailed Study portion of the ERCDC program may be succinctly stated as: (a) The underground facilities were to be designed to protect against a severe reactor meltdown accident -- without being unduly influenced by expectations with respect to the improbability of the accident; (b) Public Safety must represent the primary design goal of the engineered Accident Management Systems for fission-product control for the conceptual underground facilities; (c) After consideration of public safety, a conscious effort must be made to achieve a facility design of reasonably low cost -- though tradeoffs between operational convenience, maintainability, and in-plant safety must also be considered; (d) The Accident Management System designs must be developed with the recognition that they will be compared against a standard of total, passive protection to the public from the results of a broad range of postulated severe accidents -- that is, wherever possible, designs for the Accident Management System should be demonstrably effective against perceived catastrophic nuclear power plant accidents; (e) Thoroughly credible facility designs and analyses of public consequences must be developed, based upon documented assumptions and well-defined engineering methods.

The specifications for the Public Consequence Analysis study and the Facility Design and Cost Estimate programs were selected with the basic goal of providing realistic conditions for the investigations. While it was not considered desirable to underestimate accident conditions or design requirements, neither was excessive conservatism sought. Baseline specifications were particularly chosen with the concept of realism in mind. Sensitivity studies were selected so that the parameters specifying the conditions for the studies were expected to establish reasonable bounds of the design and consequence analyses.

As the final output from the Detailed Study portion of the ERCDC program, a set of realistic designs and cost estimates for underground facilities was sought which would be based upon reasonable conditions for construction in California. Similarly the consequence analyses to be performed were expected to be at least generically representative of California siting conditions. These goals dominated the exercise of developing the program plans, requirements, and specifications for the Detailed Study Phase of the program.

7.0 CONCLUSIONS AND RECOMMENDATIONS

This study was initiated with two basic goals. The first was to investigate the designs of underground nuclear power plants, to scope their design requirements, (in terms of both structural parameters and concept effectiveness in mitigating the results of severe core-meltdown reactor accidents) and assess the state-of-the-art for design and cost methodology. The second goal was to develop program plans, including work statements, program requirements and contractual specifications for conducting more detailed studies of (a) underground nuclear power plant designs and associated costs and (b) the effectiveness of underground plants in reducing the consequences of severe reactor meltdown accidents. The results of this study were intended to provide a basis for significant decisions with respect to both the general aspects of the technological feasibility effectiveness of underground nuclear power plant construction, and the scope of more detailed studies to be conducted subsequently. Therefore objectivity and realism were constant considerations throughout the study, in the effort to develop both a factually accurate and California specific study.

This section of the report summarizes the findings of the first-half of The Aerospace Corporation study and discusses the implications of the results for structuring the second-half (Detailed Study) portion of the ERCDC program.

7.1 THE EXPERIENCE BASE FOR DESIGN

A study was made of both existing underground nuclear power plants and previously conducted studies of these types of facilities to determine the degree to which relevant information already existed. It was found that although there was some relevant construction experience available from four European underground nuclear power plants and a relatively large number of hydro-power facilities around the world, none of these plants had been designed with the specific goal of minimizing consequences from severe reactor meltdown accidents. Thus, none of the facility designs were

directly pertinent to the principal objectives of this study, although the hydro-electric plant construction experience did substantiate the feasibility of excavating large underground chambers in rock. Moreover, cost data were not available for the underground nuclear facilities. Even if the data had been available, they would not have been directly applicable due to the dissimilarities of the design objectives for the facilities.

A review of the current literature dealing with design studies of underground nuclear power plants showed, similarly, that most of the studies were not specifically directed towards control or minimization of the consequences of severe (Class-9) reactor meltdown accidents.¹ Some attempts have been made to estimate costs of underground nuclear facilities. However, in some of these cases, adequate levels of design had not been developed to support the cost estimates; while in others, details of the cost estimates were not available for evaluation during the course of this study.

Consequently, neither entirely appropriate construction experience nor design studies were found for sources of underground concept designs, reliable estimates of costs of construction, or analyses of the effectiveness of such facilities in reducing the public consequences of (Class-9 reactor) accidents. Thus, it was essential to conduct some bounding analyses of underground facility design requirements before concepts could be evaluated or program plans developed for more detailed studies. The principal results of these analyses are presented below.

¹Subsequent to the completion of the first-half of the Aerospace program several documents have been published (e.g., studies by Sandia, Albuquerque (Ref. 7.1) and West German sources (Ref. 7.2) which have addressed Class-9 accident consequences at least indirectly. These sources were not available for review during this study. However, they appear to have pertinent results which will be analyzed and evaluated during the Detailed Study portion of the ERCDC program.

7.2 MITIGATION OF ACCIDENT CONSEQUENCES

Accident consequences were reviewed and analyzed. Three dominant modes of reactor containment structure failure were identified:

(a) penetration of containment structures by massive, high-velocity missiles launched by steam explosions in the reactor pressure vessel, (b) rupture of containment structures due to development of excessive pressures (overpressurization) from the confined energy of the radioactively decaying fission products released in the course of the accident, (c) meltthrough of the foundation base mat of the containment structure by the molten core material. Though the probability of any of these containment failure mechanisms occurring has been calculated as being very small¹ in an absolute sense (Ref. 7-3), on a comparative basis the relative likelihood of failure by the overpressurization mode is far greater than the other two means of loss of containment integrity cited -- at least for surface-sited nuclear power plants.

If a Class-9 accident occurs, containment failure may occur in a short time span following the accident. For surface sited plants, failures induced by steam explosions or overpressurization may occur in a few hours after the core meltdown accident, while meltthrough is associated with failure processes occurring over periods on the order of one day (Ref. 7-4).

¹There is much debate within the scientific community, at present, with respect to the precise numerical values of the calculated probabilities of such events occurring and even whether such calculations can be realistically performed in an absolute framework. A thorough discussion of this topic is beyond the scope of this presentation. However, irrespective of the specific values debated for the failure probabilities, there is general agreement that they are small. Moreover, only the relative values are significant for this discussion.

In an underground facility, the consequences described above for the failure modes of surface-sited plants are altered and probably diminished as a result of the underground environment. The possibility of breaching the containment by steam explosion-launched projectiles is evidently sharply reduced. Failure by meltthrough could still be a non-trivial problem; however, the path lengths for fission product transport and release to the atmosphere are obviously lengthened (when compared to path lengths for surface facilities) by amounts at least equivalent to the depth of burial of the underground reactor containment structure.

Failure by overpressurization was judged to be the dominant and potentially most serious problem for underground facilities. Analyses showed (cf Section 3.1) that an analogy can be drawn between the behavior of an underground containment structure and an oversized defective household pressure-cooker. Under accident conditions within a sealed underground facility, the decay heat of the fission products could be confined within the structure until destructive pressures were reached which the surrounding rock or soil might be incapable of withstanding. This behavior has been vernacularly referred to as a "pressure-cooker effect."

In the event of destructive overpressurization of an underground facility, the potential for development of large fissures in the earth media was a source of substantial concern throughout the course of these studies. Such fissures could provide direct paths for escape of released fission products from a failed underground structure to the surface. To reduce the probability of catastrophic fissure development, accident management systems were conceived and defined (cf Section 6.1) which would provide engineered methods for limiting pressures developed in the containment structures in the course of a Class-9 accident. The accident management systems thus defined were intended to provide engineered pathways for release of the fission products into expansion volumes/ filter beds, in order to entrap fission products and increase their transport times to the surface, so that levels of radioactivity released to the environment would be greatly diminished.

Preliminary estimates of the effectiveness of underground facilities in diminishing the releases of radioactive fission products in the event of an accident (cf Section 3.1) indicate that large reductions can be achieved for even the most mobile fission products -- i.e., the noble gases, such as Kr-85. In the case of soluble or particulate fission products, diffusion through the earth media was found to produce even more dramatic reductions in total releases of radioactivity. An example calculation was made for Sr-90 which showed reductions in radioactive releases by nearly three orders of magnitude were obtained after diffusion through only 10 m of soil. Both noble gases and soluble particulate forms of fission products were shown to be delayed in time of release by large factors when compared to similar postulated accidents in surface plants. Peak release rates for Kr-85 were calculated to occur about one year after meltdown, and the total release was calculated to span about 60 years. Similarly, measurable radioactivity releases for SR-90 were shown to occur only after a period of nearly 70 years, (about 3 half-life periods for the radioactive isotope) with continuing releases at very low rates for about an additional 100 years.

The contaminated area at the surface, where significant radioactivity was found to be transmitted through the soil from the underground release point, was shown to be very small. For release depths as near the surface as 10 m, a circular area with a diameter smaller than 100 m (about the length of a football field) would encompass the region where damaging radioactivity might be obtained from fission products migrating to the surface.

Thus, if the development of new fissures could be prevented in the underground medium, which might provide direct pathways to the surface for fission products, underground construction of nuclear power plants appeared to provide significant protection from releases of radioactivity.

7.3 STATE GEOLOGICAL AND CLIMATOLOGICAL ASSESSMENT

A brief survey was conducted of the topographical, geological, and climatological attributes of the State of California pertinent to technically acceptable underground siting of facilities. After conducting a first-order map survey of the State (cf Section 3.2) the Central Valley and East Mojave regions were identified as areas of appropriate geology for both soil and rock sites for underground facilities. Climatological patterns of the Central Valley and East Mojave regions of the State appeared to be reasonably uniform as well. Thus, these two regions were recommended for further evaluation in the subsequent Detailed Study phase of the second half of the ERCDC program.

7.4 REDUCTIONS IN SEISMIC MOTIONS WITH DEPTH

To assess the impact of the generally severe seismic environment of the State on the feasibility of constructing underground nuclear power plants, estimates were made of seismic motions and structural loads for subsurface facilities. The investigation also included an evaluation of the applicability of current seismic design methods for surface nuclear facilities to underground design. Pertinent empirical evidence available on underground measurements of seismic motions was reviewed and evaluated. The data support the inference of decreasing levels of motion with increasing depth beneath the surface. However, when compared to typical California seismic design requirements, most measurements have been obtained from relatively weak earthquake motions, with recorded accelerations generally substantially less than 0.1 g. To overcome the limitations of the empirical data for larger seismic input motions, finite-element calculations were performed for free-field motions and structure-medium interactions in a "typical" Central Valley site subject to 0.5 g seismic acceleration inputs.

Based upon the combined results of both calculations and empirical data, reduction factors were defined for application as preliminary design criteria. The results indicated (cf Section 3.3, Table 3-9) that peak accelerations at depths of 30 m (or greater) would be reduced by a factor of 0.5 in soils and 0.67 in rock sites. Factors for peak values of velocity and displacement were also shown to be reduced, although the specified peak displacement was only reduced by a factor of 0.83 for soils. No reduction in displacement was recommended for rock sites.

Structure-medium interactions were also shown to be reduced at modest burial depths, e.g., about the depths for berm-containment designs. Horizontal motions at the base of such buried structures were found to be reduced by a factor of 0.75, when compared to a surface facility at frequencies of interest for structural design. At the same frequencies, horizontal motions at the roof line of buried structures would be reduced by larger factors (about 0.4) when compared to equivalent motions at the top of a containment structure for a surface-sited facility. Vertical structure motions, on the other hand, were found to be only modestly reduced (by about a factor of 0.9) for these relatively shallow depths of burial, when compared to similar surface-sited structures.

The apparent reductions in input and structural motions for the underground facility might be sufficient to permit corresponding economies in amounts of concrete and steel in structures, as well as the use of standardized plant equipment developed for use in more seismically benign areas of the U.S. These factors suggest that some cost advantages may occur in underground design due to reduced seismic requirements. Whether these would be sufficient to offset other factors adversely penalizing underground construction costs (such as added excavation costs and required increases in wall thickness to resist the earth loads associated with underground construction) will be better determined after the Detailed Study phase of the ERCDC program is completed.

7.5 DESIGN AND COST IMPACTS OF UNDERGROUND CONSTRUCTION

A number of underground construction design concepts were postulated and analyzed. For berm-contained concepts, three basic design variations were considered: (a) cut-and-cover concepts (with excavation depths approximately equal to maximum structure heights), (b) balanced cut-and-cover (using shallower excavation depths), and (c) cover-only concepts. For mined-rock cavity concepts, hillside and vertical entry design variations were evaluated. Rock properties were examined with regard to their influence on underground structure design and construction. It was concluded that relatively high quality igneous and metamorphic rocks, such as granite, quartzite, or gneiss, would have the most satisfactory properties for underground construction. Comparative estimates of nominal excavation requirements for bermed and mined-rock facilities were made on the basis of the idealized design concepts developed. No severe technological problems with construction of underground facilities in either soil or rock were found in the study. However, free-standing rock excavations with spans in excess of 100 ft are rarely seen in contemporary design practice. Underground nuclear power plants will require spans of this order to house the nuclear steam supply system, unless major modifications were to be made to this important system. Thus, design and construction of rock cavities for underground nuclear power plants will approach the state-of-the-art of current practices, and estimates of their construction costs are therefore subject to some uncertainty.

7.5.1 Selection of Facility Systems for Underground Emplacement

A functional systems analysis was made of surface nuclear power plants. Based upon several alternative sets of criteria for functional selection of reactor systems to be placed underground, seven different combinations of hybrid facilities (having some elements at the surface and others underground) were evaluated.

A simplified analysis was conducted of construction practices for these seven functionally-derived, hybrid facility concepts in mined-rock designs, in order to estimate relative differential costs. Contrary to initial intuition, those configurations where the majority of the plant systems were placed underground (including the reactor, turbine-generator, and related auxiliary functions) were found to be relatively less expensive than undergrounding only the nuclear steam supply (reactor) related systems. In the latter cases, cost differences (compared to surface plants) were estimated to be increased by factors of about 3 or 4. These large differentials were primarily due to the costs of long lengths of high-pressure, high temperature steam and/or reactor coolant lines required for connections between systems within the underground structures and the steam generators or turbine-generators (and, in some cases, other related reactor systems) located on the surface. A second major contribution was attributable to the need to provide radioactive fission product containment structures at both the surface and subsurface locations when elements of the nuclear steam supply system were divided between the two locations.

7.5.2 Cost Uncertainties Related to Facility Design

Factors were examined which contribute to uncertainties in estimates of the costs of underground facilities when designed at the conceptual level. Three categories of factors contributing to differences between initial estimates and actual costs of plants were reviewed. These were: (a) factors related to prices of goods, services, and money, i.e., interest and escalation rates, (b) direct errors in estimation of plant construction costs or duration of the construction period as defined by initial plans, design drawings and schedules for the facilities, and (c) changes in plant design or construction schedules required after initiation of construction and/or preparation of initial cost estimates.

In order to determine significant areas for attention in conducting the A/E design efforts during the Detailed Study portion of the ERCDC program, work was concentrated on evaluation of cost uncertainties resulting from the second category of factors, direct design errors. Inexperience in the development of costs for new design concepts was found to be the dominant contributing factor within this category of cost uncertainties. Inexperience contributes to problems such as: (a) design errors, omissions, and ambiguities, (b) uncertainties over effects of restricted access to work space, and (c) modifications to established construction methods. These factors are difficult to evaluate during conceptual design and planning activities.

Based upon data gleaned from many industries, estimates of cost uncertainties associated with inexperience were developed. On this basis alone, engineering cost estimates were judged to be uncertain (in the direction of underestimation) by about 15 percent. This uncertainty appeared to be dominated by excavation and underground structural design related factors. Other significant, but relatively smaller contributions to the cost uncertainty are related to high-pressure, high-temperature piping, such as steam and feedwater piping systems.

7.5.3 Relative Evaluation of Underground Design Concepts

Finally, semi-quantitative evaluations were made of several berm and mined-rock candidate concepts. Factors considered in the relative evaluation included: (a) earth handling requirements and costs, (b) siting area requirements, (c) potential availability of underground siting opportunities in California, (d) equipment hardware and facility design impacts, and (e) aesthetic impacts. Overall, the berm-contained and mined-rock concepts appeared to be essentially indistinguishable in relative costs and impacts.

7.6 CONCEPTS FOR AUGMENTING SAFETY OF SURFACE NUCLEAR POWER PLANTS

To provide perspective on the relative technological feasibility and costs of underground plants, a brief review was made of measures which could be taken to augment the safety of surface plants of current designs against Class-9 accidents (cf Section 5). Although offshore and remote siting were included in the reviews as alternatives to underground siting, the analysis was primarily concentrated on potential modifications to LWR systems to increase their capability for control of severe core-meltdown accidents, thereby preventing (or greatly reducing) escape of radioactivity into the biosphere.

A systems approach was taken to analyze several concepts for Class-9 accident mitigation with respect to their capability to: (a) limit steam explosions (b) prevent penetration of the containment boundary by the molten-core, (c) prevent vapors produced by the meltdown processes from overpressurizing the containment structure to failure, and (c) provide long-term heat removal of the fission-product decay heat from within the containment boundary. A variety of subsystems which could provide these four general capabilities appear to be within the limits of current technology. These include such concepts as blast shields for protection against steam explosions; core-catchers to prevent the molten core from penetrating the containment; filtered venting concepts to prevent excessive pressure buildup within the containment; as well as large, elaborate heat-sink concepts to provide for decay heat removal.

Synthesizing systems from components which had been proposed in the open literature by researchers, four concepts were analyzed which proffered the capability of solving the problem areas cited above. In two of the concepts, designs were proposed to provide complete containment of all fission-products released in an accident, through the use of core-catchers and large-capacity heat-sinks. The first concept proposed the use of a massive melting bed as a dry core-

catcher and a long-term heat removal mechanism. The second concept incorporated a liquid-cooled core-catcher which maintained a steam-filled atmosphere within a free-standing steel containment structure. In this concept, the steel containment structure would be immersed within a large lake, which would act as a heat-sink by maintaining a relatively low temperature in the steel shell of the containment, thus inducing condensation of the steam at the interior surface. This process would also limit the pressures and temperatures within the containment structures to the corresponding saturation conditions for steam. A third concept was based upon pressure-relief for the vapors within the containment through a controlled and filtered venting system. In the fourth concept, a patented system design for meltdown prevention was reviewed. This concept was taken as an example of advanced meltdown prevention systems and was not intended to represent a position of advocacy of the specific system.

On the basis of a qualitative evaluation of the four system concepts analyzed, it was concluded that complete control (or reliable prevention) of core-meltdowns was feasible for surface-sited facilities, despite technological uncertainties which were apparent in the evaluation of each of the systems. The technological problems appear to be resolvable since the major phenomena that could lead to containment violation (heat, gas, and missile production) are relatively simple physical processes which can be analyzed and are subject to engineering design solutions. With sufficient thermal capacity to accommodate heat production and volume to accommodate gas and vapor production, meltdowns appear to be amenable to control through conventional applied engineering methods.

The question of relative costs of such methods for surface facility augmentation vs. underground construction was beyond the scope of this portion of the study. Similarly, the general question of whether such methods are cost-effective is a larger problem, which will be addressed in somewhat greater detail in the second-half of the ERCDC study.

7.7

DEVELOPMENT OF SPECIFIC PROGRAM PLANS FOR
THE DETAILED STUDY PHASE OF THE ERCDC STUDY

Section 6 of this report describes the background and approach taken toward development of detailed statements of work, contract specifications, and program plans for more detailed follow-on studies to be conducted under ERCDC sponsorship following the Initial Program Definition Studies presented herein. The plans for the follow-on studies were developed in accordance with the following broad guidelines:

- a. The underground facilities were to be designed to protect against a severe reactor meltdown accident -- without first order regard for the improbability of the accident.
- b. Public safety must represent the primary design goal for engineered Accident Management Systems for fission-product control for the conceptual underground facilities.
- c. After consideration of public safety, a conscious effort must be made to achieve a facility design of reasonably low cost -- although trade-offs between operational convenience, maintainability, and in-plant safety must also be considered.
- d. Accident Management System designs must be developed with the recognition that they will be compared against a standard of total, passive protection to the public from the results of a broad range of postulated severe accidents -- that is, where possible, designs for Accident Management Systems should be demonstrably effective against perceived catastrophic nuclear power plant accidents.

- e. Thoroughly credible facility designs and analyses of public consequences must be developed, based upon documented assumptions and well-defined engineering methods.

The desire for realistic designs, schedules, and cost estimates was another factor influencing formulation of the contractual documents developed for the follow-on studies. Similarly, results of public consequence analyses were expected to be reasonably representative of the demographic and geophysical conditions of California. To this end, contractual specifications were selected with the broad objective of ensuring the development of both reasonable and conservative designs for which cost estimates could be derived which were neither excessively optimistic nor pessimistic. Again, it was recognized that much of the uncertainty in cost estimates associated with conceptual designs is contributed by schedule uncertainties. Consequently, designers were tasked with development of realistic schedules, with a representative range of reasonable upper and lower estimates of schedule perturbations. Demographic and siting specifications were based upon the results described in Section 3.2 of this report.

In conjunction with the ERCDC staff, guidelines were developed to define a series of ascending levels of public safety in designs for Accident Management Systems. Five levels were defined (cf Section 6.1.1) ranging from a Design Level Zero concept, based simply on underground emplacement of a nuclear power plant, with essentially the same design features as a surface facility, to an underground concept (Design Level 4) providing for complete isolation from the biosphere (within an essentially leak tight containment structure) of any fission-products released during a meltdown accident. The Design Level 2 concept was chosen as a baseline for follow-on studies. This Accident Management System utilizes the natural fission-product filtering and retention properties

of the earth by providing an engineered pressure relief mechanism from the containment to an earthen fission product release zone outside of the walls of the facility. Design Levels 1 and 3 were developed for analysis in sensitivity studies to be conducted in the ERCDC follow-on program. The Design Level 4 concept was not included within the scopes-of-work for follow-on contracts, due to limitations in time and funding for the overall ERCDC program.

In order to ensure the relevance of analyses of public consequences to be performed during the Detailed Study portion of the program, the specifications to the contractor statement of work were selected to be essentially identical to the specifications used in the A&E design contracts. However, requirements were imposed for developing supplementary demographic and meteorological conditions for representative regions within the State.

The scope of the study was constrained to ensure that the contractor would concentrate on providing estimates of public consequences and environmental impacts rather than attempting to develop estimates of accident probabilities associated with the underground facilities. It was recognized that this approach limited the capability to provide statistical estimates of underground nuclear power plant risks. However, reliable estimates of accident probabilities cannot be made without considerably more facility design detail than is to be provided by the conceptual designs developed in the Detailed Study portion of the program. However, development of relative consequence estimates did appear to be a reasonable goal for the study. These estimates of consequences would permit evaluation of the effectiveness of the concepts, while realistic cost estimates would provide a similar basis for evaluation of the economic feasibility of underground nuclear power plant concepts. Together with other investigations being conducted to evaluate the question of the necessity for the underground facilities, the planned studies were judged to provide a reasonably comprehensive program for satisfying the requirements of the legislatively mandated study.

REFERENCES

- 7-1 Allensworth, James A., Finger, John T., et al, "Underground Siting of Nuclear Power Plants: Potential Benefits and Penalties," SAND76-0412/NUREG-0255, Sandia Laboratories, Albuquerque, New Mexico, August 1977.
- 7-2 KFA, "Beurteilung der unterirdischen Errichtung von Kernkraftwerken im Boden in einer offenen Baugrube," Kernforschungsanlage Julich GmbH, June 1977.
- 7-3 U.S. NRC, "Reactor Safety Study," WASH-1400, Main Report, p. 79, October 1975.
- 7-4 Ibid., p. 78.

APPENDIX I
SYSTEM DESIGN AND PUBLIC CONSEQUENCE
ANALYSIS SPECIFICATIONS

Three major contracts are planned for the Detailed Study portion of the second-half of the ERCDC program of investigations of underground nuclear power plant siting. Two major architect-engineers (A/E) will develop design and cost data for (a) berm-contained and (b) mined-rock underground facility concepts respectively. A third major contract is planned for conducting analyses of potential public consequences from postulated severe accidents in comparable surface and underground facilities. The specifications presented in Appendices I-1, I-2, and I-3 were generated for the purpose of defining the scope of the contractual obligations of the three contractors in conducting the detailed design and public consequences studies for ERCDC under the technical direction of The Aerospace Corporation. The specifications were derived from the results of preliminary analyses conducted by The Aerospace Corporation. Final specifications will represent the integrated results of recommended modifications provided by The Aerospace Corporation, ERCDC, A/E contractors, and the Public Consequence contractor.

Appendices I-1 and I-2 describe specifications for the development of designs and cost data for a berm-contained nuclear power plant and a mined-rock facility, respectively. Appendix I-3 gives the specifications for analyses of consequences to the public from a series of postulated Class-9 accidents.

APPENDIX I-1

Table I-1. Specifications for a Study of Buried, Berm-Containment Design Concepts for Underground Nuclear Power Plants

	Baseline	Sensitivity
I. <u>Power Plant Characteristics</u>		
Reactor and Thermal Capacity		
Westinghouse RESAR 41 (3800 MW)	X	
General Electric GESSAR (3800 MW)		(1)
Babcock & Wilcox Standard 241 (3800 MW)		(2)
Combustion Engineering CESSAR (3800 MW)		(2)
Plant Electrical Capacity, 1300 MW Net	X	NA
Cooling Water Supply Concept - Forced Convection	X	NA
Number of Units Sharing Common Facility	Single	NA
Nuclear Steam Supply Configuration		
o Standard Vendor Configuration	RESAR 41	(1) GESSAR
o Alternative NSSS Configuration to Accommodate Dimensional Constraints of Buried Containment	NA	NA
II. <u>System/Component Locations</u>		
Nuclear Steam Supply System	Subsurface	NA
Waste Processing and Storage	Subsurface	NA
Turbine Generator	Subsurface	NA
Main Transformer	Surface	NA
Spent and New Fuel Storage and Handling	Surface**	NA
Control Center	Surface	NA
Emergency Diesel Generators	Surface	NA

Table I-1. (Continued)

	Baseline	Sensitivity
III. <u>Siting Data</u>		
Soil Description		
Variation with Depth	Soil Model 1	Soil Model 2
Soil Moisture	Moist	Dry
Groundwater Specification		
Facility Location with Respect to Water Table	Above	Within
Seismic Criteria - Surface Peak Acceleration ⁽³⁾	0.5 g	a) 0.67 g
Rock Characteristics	NA	Soil a) Model 3
Site Remoteness	Central Valley Rural (TBD)	East Desert Remote(TBD)
Meteorological Conditions	Humid (TBD)	Dry (TBD)
IV. <u>Fission Product Containment Requirements</u>		
Pressure-Time Relationships	TBD ^{***}	B/L
Energy-Time Relationships	TBD	B/L
Pressure Management Design Concepts	Level 2	a) Level 1 b) Level 3
Minimum Depth of Containment Cover	50 ft.	a) 25 ft. b) 100 ft.

Table I-1. (Continued)

	Baseline	Sensitivity
Surface/Underground Structure Interface Relationships		
Allowable Leakage Rates	TBD	B/L
V. <u>Schedule Variations</u>	Nominal	a) Short b) Long
VI. <u>Reference Surface Power Plant Characteristics</u>		
Power Plant Characteristics	B/L (or equivalent)	NA
System/Component Locations	Standard or Replicated Surface Facility B/L	NA
Siting Data	B/L	NA
Fission Product Containment	Class 8 (DBA)	Level 3

- (1) BWR concept is a major design alternative for which plant layout must be prepared.
- (2) No plant layouts are required.
 - * B/L = Baseline. (Soil Model 1)
 - ** Minimal fuel handling system to be provided within underground portion of facility. Full scale spent fuel storage pit at surface.
 - *** TBD - (To be determined).
- (3) Subsurface seismic criteria (TBD).

Table I-1. Soil Model No. 1 of Appendix I-1 (Continued)

Table I-1. Soil Model No. 2 of Appendix I-1 (Continued)

Depth (Feet)	Well Log		Soil Description
		Classification	
0		ML	Fine silty sand; density 108 lb/ft ³ ; water content 65%; relative density 65%; bearing capacity 850 PSF; permeability (K) = 3×10^{-3} cm/sec.;
20			
40		CL	Clay, slightly sandy; low plasticity; density 120 lb/ft ³ ; water content 31%; unconfined compressive strength 1200 PSF; liquid limit $W_L = 43\%$; plastic limit $W_L = 26\%$; plastic index = 17%; permeability (K) = 10^{-4} cm/sec., angle of internal friction 40°;
60			
80			
100			Coarse sand, poorly graded; density 128 lb/ft ³ ; water content 80%; bearing capacity = 7200 PSF; permeability (K) 1.8×10^{-1} cm/sec.;
120		SP	
140			
160			
180			Coarse, sandy gravel; occasional rocks to 18" diameter; density 134 lb/ft ³ water content 85%; relative density 100%; bearing capacity 8.5 TSF; permeability (K) = 1.4 cm/sec.
200			
220			
240			
260			
280		GP	
300			
320			
340			
360			

Table I-1. Soil Model No. 3 of Appendix I-1 (Continued)

<u>Depth</u>	<u>Description</u>
0-10 feet	Same material as upper layer of Soil Model No. 1
10-50 feet	<p>Material - shale</p> <p>Dip of bedding planes - 45° from horizontal</p> <p>Weathering - highly and fractionally broken</p> <p>Unconfined compressive strength - 6500 psi</p> <p>Youngs modulus - 0.8×10^6 psi</p> <p>Poissons ratio - 0.25</p> <p>Density - 145 lbs/ft³</p> <p>Permeability - 10^{-3} cm/sec</p> <p>Porosity - 7%</p>
50 feet and below	<p>Material - shale</p> <p>Dip of bedding planes - 45° from horizontal</p> <p>Weathering - slight with tight laminations and unaltered transverse fractures from 3-4 feet apart</p> <p>Unconfined compressive strength - 14,000 psi</p> <p>Youngs modulus - 1.4×10^6 psi</p> <p>Poissons ratio - 0.25</p> <p>Density - 155 lbs/ft³</p> <p>Permeability - 10^{-6} cm/sec</p> <p>Porosity - 1.2%</p>

APPENDIX I-2

Table I-2. Specifications for a Study of Mined Underground Design
Concepts for Underground Nuclear Power Plants

	Baseline	Sensitivity
I. <u>Power Plant Characteristics</u>		
Reactor and Thermal Capacity		
Westinghouse RESAR 41 (3800 MW)	X	
General Electric GESSAR (3800 MW)		(1)
Babcock & Wilcox Standard 241 (3800 MW)		(2)
Combustion Engineering CESSAR (3800 MW)		(2)
Plant Electrical Capacity, 1300 MW Net	X	NA
Cooling Water Supply Concept - Forced Convection	X	NA
Number of Units Sharing Common Facility	Single	NA
Nuclear Steam Supply Configuration		
o Standard Vendor Configuration	RESAR 41	(1) GESSAR
o Alternative NSSS Configuration to Accommodate Dimensional Constraints Underground Containment	NA	YES
II. <u>System/Component Locations</u>		
Nuclear Steam Supply System	Subsurface	NA
Waste Processing and Storage	Subsurface	NA
Turbine Generator and Main Transformer	Subsurface	Surface
Spent and New Fuel Storage and Handling	Surface**	NA
Control Center	Surface	NA
Emergency Diesel Generators	Surface	NA

Table I-2. (Continued)

	Baseline	Sensitivity
III. <u>Siting Data</u>		
Rock Description	Rock Model 1	a) Rock Model 2 b) Rock Model 3
Groundwater Specifications	Dry	Dewater (TBD) ***
Seismic Criteria - Surface Peak Acceleration (4)	0.5g	a) 0.3g b) 0.67g
Topography	Hillside (20% Slope)	Flat
Site Remoteness	Central Valley Rural (TBD)	East Desert Remote (TBD)
Meteorological Conditions	Humid (TBD)	Dry (TBD)
IV. <u>Fission Product Containment Requirements</u>		
Pressure-Time Relationships	TBD	Baseline
Energy-Time Relationships	TBD	Baseline
Pressure Management Design Concepts	Level 2	a) Level 1 b) Level 3
Minimum Depth of Containment Cover	50 ft. (5)	1000 ft.

Table I-2. (Continued)

Surface/Underground Structure Interface Relationships	Baseline	Sensitivity
Allowable Leakage Rates	TBD	NA
V. <u>Schedule Variations</u>	Nominal	a) Short b) Long
VI. <u>Reference Surface Power Plant Characteristics</u>		
Power Plant Characteristics	Baseline (or Equivalent)	NA
System/Component Locations	Standard or Replicated Surface Facility	NA
Siting Data	(3)	NA
Fission Product Containment	Class 8 (DBA)	NA

(1) BWR concept is a major design alternative for which a plant layout must be prepared.

(2) No plant layouts are required.

* NA = Not applicable.

** Minimal fuel handling system to be provided within underground portion of facility. Full scale spent fuel storage pit at surface.

*** TBD - (To be determined).

(3) Site Geology: Baseline-Soil (Soil Model 1)

Groundwater: NA

Seismic: 0.5g

Remoteness: Central Valley Rural (TBD).

Meteorological: Humid relative humidity (TBD); Wet bulb temp. (TBD).

(4) Subsurface seismic criteria (TBD).

(5) Use optimal construction depth, if greater than 50 ft.

Table I-2. (Continued)

ROCK MODEL NO. 1

Overburden: 38' deep consists of decomposed granodiorite.

Rock Mass: Coring to 1,000' revealed the following: Hard, dense, coarse grained, light gray granodiorite. The rock mass has been cut by two small diatase dikes and five very small aplite dikes. They are not zones of weakness.

Seismic survey shows 1,100/22,000 ft/sec interface at 65 ft which correlates with the depth of granodiorite weathering.

Material Properties:

The rock properties from 65 ft to the total depth are relatively uniform and yield the following average properties:

Specific gravity	2.77
Percent porosity	0.73
Schmidt rebound no.	48.5
Compressive strength, psi	22,470
Young's Modulus, psi x 10 ⁶	6.5
Tensile strength, psi	1,030

Fracture Frequency:

38'-65' highly fractured, weathered, FF≈4" at 38' grading to 12" at 65' RQD = 52

65'-91' moderately fractured; FF≈18" RQD = 75:

91'-1,000' moderately fractured; FF≈30" avg RQD = 97

Tectonic Stress Field:

38'-91' average horizontal stress is 70% of the vertical stress:

91'-1,000' average horizontal stress is 120% of the vertical stress.

Table I-2. (Continued)

ROCK MODEL NO. 2:

Overburden: 13 ft deep and consists of well-graded sand, gravel and occasional rocks of boulder size.

Rock Mass: Coring to 1,000' revealed the following: hard dense, fine to medium grained medium gray diabase gabbro.

The uphole seismic survey shows 1,200/16,500 ft/sec interface at 21 ft deep which corresponds to the depth of gabbro weathering.

Material Properties:

The rock properties from 21 ft to 1,000 ft are comparatively uniform and yield the following average properties:

Specific gravity	3.010
Percent porosity	0.30
Schmidt rebound no.	42
Compressive strength, psi	<47,500
Young's modulus, psi x 10 ⁶	8.6
Tensile	2,170

Fracture Frequency:

21'-111' moderately fractured; FF≈16" RQD = 86
111' - 1,000' relatively intact; FF≈30" RQD ≈ 100

Tectonic Stress Field:

21'-111' average horizontal stresses 60% of vertical
111'-1,000' average horizontal stresses are about 150% vertical stress.

Table I-2. (Continued)

ROCK MODEL No. 3:

Overburden: 63 ft of silts, sandy gravels and boulders: occasional outcrops of base material throughout area.

Rock Mass: Fine crystalline salmon red granite. The rock mass is highly fractured and evidence of weathering and alteration exists to a depth of 145 ft.

Uphole seismic velocity surveys show 1,200'/9,000'/sec interface at 63 ft and 9,000'/14,400'/sec interface at 145 ft which corresponds to the overburden and weathering depths described above.

Material Properties: The rock properties determined for the rock mass to the total depth of 1,000 ft average as follows:

<u>Property</u>	<u>63'-145'</u>	<u>145'-498'</u>	<u>498'-1,000'</u>
Specific gravity	2.637	2.648	2.665
Percent porosity	1.45	0.87	0.45
Schmidt rebound no.	48.3	-	56.8
Compressive strength, psi	7,600	11,700	26,600
Young's modulus psi x 10 ⁶	7.3	8.0	10.0
Tensile strength, psi	640	870	1,230

Fracture Frequency: 63'-145'; highly fractured and weathered; fracture frequency grades from $\approx 3''$ to $\approx 6''$ at 145'. RQD = 58%

145'-1,000' - average fracture frequency $\approx 6''$. RQD = 71%.

Tectonic Stress: 63'-145'; average horizontal stresses $\approx 40\%$ of vertical stresses.
145'-498'; average horizontal stress $\approx 60\%$ of vertical stresses.
498'-1,000'; average horizontal stress $\approx 110\%$ of vertical stresses.

Table I-2. (Continued)

APPENDIX I-3

Table I-3. Specifications for Analysis of Public Consequences from Postulated Severe Accident Sequences in Underground Nuclear Power Plants

	<u>Baseline</u>	<u>Sensitivity</u>
<u>I. Power Plant Characteristics</u>		
<u>Reactor Type and Thermal Capacity</u>		
Westinghouse RESAR	X	
General Electric GESSAR (3800 MW)		X
<u>II. Plant Configurations and Functional Parameters</u>		
Layouts (Berm Containment and Underground Concepts)	Level 2	a) Level 1 b) Level 3
Maximum Design Leakage Rate (Unimpaired)	0.1%/day	B/L
Minimum Depth of Cover		
Soil Site	50 ft.	a) 25 ft. b) 100 ft.
Rock Site	200 ft.	a) 1000 ft.
<u>III. Siting Data</u>		
<u>Site Material Description</u>		
Soil Site (Fill Characteristics - TBD)	Soil Model 1	a) Soil Model 2 b) Soil Model 3 Rock
Rock Site	Rock Model 1	a) Model 2 b) Rock Model 3
<u>Groundwater (TBD)</u>		
Soil Site (Location with respect to Water Table)	Above	Within
Rock Site	Dry	Wet
Site Remoteness	By 4 Geographical Zones	Remote
Meteorological Conditions		
	Rural	

Table I-3. (Continued)

	<u>Baseline</u>	<u>Sensitivity</u>
IV. <u>Conditions at Time of Accident</u>		
Decay Heat	ANS + 20%	
Radionuclide Inventory (Contractor Developed)	Most Severe	
V. <u>Reference Surface Power Plant Description</u>		
Power Plant Characteristics	B/L *	
Containment/BOP Layouts	Standardized Plants	
Site Characteristics (By 4 Geographical Zones)		
• Land Use	B/L	
• Remoteness	B/L	
• Meteorological Conditions	B/L	
Conditions at Time of Accident		
• Decay Heat	B/L	
• Radionuclide Inventory	B/L	
• Containment Failure Mode	PWR-2 (WASH-1400)	

* B/L = Baseline Characteristics

Table I-3 (Continued)

Soil Model No. 1

Table I-3. (Continued)

Soil Model No. 2

Depth (Feet)	Well Log Classification		Soil Description
0		ML	Fine silty sand; density 108 lb/ft ³ ; water content 65%; relative density 60%; bearing capacity 850 PSF; permeability (K) = 3×10^{-3} cm/sec.;
20			
40		CL	Clay, slightly sandy; low plasticity; density 120 lb/ft ³ ; water content 31%; unconfined compressive strength 1200 PSF; liquid limit $W_L = 43\%$; plastic limit $W_L = 26\%$; plastic index = 17%; permeability (K) = 10^{-4} cm/sec., angle of internal friction 40° ;
60			
80			
100			Coarse sand, poorly graded; density 128 lb/ft ³ ; water content 80%; bearing capacity = 7200 PSF; permeability (K) 1.8×10^{-1} cm/sec.;
120			
140		SP	
160			
180			Coarse, sandy gravel; occasional rocks to 18" diameter; density 134 lb/ft ³ ; water content 85%; relative density 100%; bearing capacity 8.5 TSF; permeability (K) = .1.4 cm/sec.
200			
220			
240			
260			
280		GP	
300			
320			
340			
360			

Table I-3. (Continued) Soil Model No. 3

<u>Depth</u>	<u>Description</u>
0-10 feet	Same material as upper layer of Soil Model No. 1
10-50 feet	<p>Material - shale</p> <p>Dip of bedding planes - 45° from horizontal</p> <p>Weathering - highly and fractionally broken</p> <p>Unconfined compressive strength - 6500 psi</p> <p>Youngs modulus - 0.8×10^{-6} psi</p> <p>Poissons ratio - 0.25</p> <p>Density - 145 lbs/ft³</p> <p>Permeability - 10^{-3} cm/sec</p> <p>Porosity - 7%</p>
50 feet and below	<p>Material - shale</p> <p>Dip of bedding planes - 45° from horizontal</p> <p>Weathering - slight with tight laminations and unaltered transverse fractures from 3-4 feet apart</p> <p>Unconfined compressive strength - 14,000 psi</p> <p>Youngs modulus - 14×10^6 psi</p> <p>Poissons ratio - 0.25</p> <p>Density - 155 lbs/ft³</p> <p>Permeability - 10^{-6} cm/sec</p> <p>Porosity - 1.2%</p>

Overburden: 38' deep consists of decomposed granodiorite.

Rock Mass: Coring to 1,000' revealed the following: Hard, dense, coarse grained, light gray granodiorite. The rock mass has been cut by two small diatase dikes and five very small aplite dikes. They are not zones of weakness.

Seismic survey shows 1,100/22,000 ft/sec interface at 65 ft which correlates with the depth of granodiorite weathering.

Material
Properties:

The rock properties from 65 ft to the total depth are relatively uniform and yield the following average properties:

Specific gravity	2.77
Percent porosity	0.73
Schmidt rebound no.	48.5
Compressive strength, psi	22,470
Young's Modulus, psi x 10 ⁶	6.5
Tensile strength, psi	1,030

Fracture
Frequency:

38'-65' highly fractured, weathered, FF≈4"
at 38' grading to 12" at 65' RQD = 52

65'-91' moderately fractured; FF≈18" RQD = 75:
91'-1,000' moderately fractured; FF≈30" avg RQD = 97

Tectonic
Stress
Field:

38'-91' average horizontal stress is 70% of the vertical stress:
91'-1,000' average horizontal stress is 120% of the vertical stress.

Overburden: 13 ft deep and consists of well-graded sand, gravel and occasional rocks of boulder size.

Rock Mass: Coring to 1,000' revealed the following: hard dense, fine to medium grained medium gray diabase gabbro.

The uphole seismic survey shows 1,200/16,500 ft/sec interface at 21 ft deep which corresponds to the depth of gabbro weathering.

Material
Properties:

The rock properties from 21 ft to 1,000 ft are comparatively uniform and yield the following average properties:

Specific gravity	3.010
Percent porosity	0.30
Schmidt rebound no.	42
Compressive strength, psi	<47,500
Young's modulus, psi $\times 10^6$	8.6
Tensile	2,170

Fracture
Frequency:

21'-111' moderately fractured; FF $\approx$ 16" RQD = 86
111' - 1,000' relatively intact; FF $\approx$ 30" RQD $\approx$ 100

Tectonic Stress
Field:

21'-111' average horizontal stresses 60% of vertical
111'-1,000' average horizontal stresses are about 150% vertical stress.

Table I-3. (Continued)

Rock Model No. 3

Overburden: 63 ft of silts, sandy gravels and boulders: occasional outcrops of base material throughout area.

Rock Mass: Fine crystalline salmon red granite. The rock mass is highly fractured and evidence of weathering and alteration exists to a depth of 145 ft.

Uphole seismic velocity surveys show 1,200'/9,000'/sec interface at 63 ft and 9,000'/14,400'/sec interface at 145 ft which corresponds to the overburden and weathering depths described above.

Material Properties:

The rock properties determined for the rock mass to the total depth of 1,000 ft average as follows:

<u>Property</u>	<u>63'-145'</u>	<u>145'-498'</u>	<u>498'-1,000'</u>
Specific gravity	2.637	2.648	2.665
Percent porosity	1.45	0.87	0.45
Schmidt rebound no.	48.3	-	56.8
Compressive strength, psi	7,600	11,700	26,600
Young's modulus psi x 10 ⁶	7.3	8.0	10.0
Tensile strength, psi	640	870	1,230

Fracture Frequency:

63'-145'; highly fractured and weathered; fracture frequency grades from $\approx 3''$ to $\approx 6''$ at 145'. RQD = 58%

145'-1,000' - average fracture frequency $\approx 6''$. RQD = 71%.

Tectonic Stress:

63'-145'; average horizontal stresses $\approx 40\%$ of vertical stresses.

145'-498'; average horizontal stress $\approx 60\%$ of vertical stresses.

498'-1,000'; average horizontal stress $\approx 110\%$ of vertical stresses.

APPENDIX II
SYSTEMS AND HARDWARE INTERFACE DATA FOR
REFERENCE PWR AND BWR NUCLEAR POWER PLANTS

The following data have been developed in order to identify reference plant interfaces between various buildings and systems components in the above ground configurations of PWR and BWR nuclear power plants. An estimate of the plant inventory of piping (quantity and size) has been provided by describing the interconnected buildings or plant areas with their associated interdependent systems.

Table II-1 contains a summary of the piping between the Stone and Webster PWR plant buildings. Information on the pipe sizes was not available from the Preliminary Safety Analysis Reports. Table II-2 contains a more detailed breakdown of the information given in Table II-1, since it relates the systems to the plant buildings and pipes involved.

Tables II-3 and II-4 also contain the summary and detailed inventory of the BWR piping inventory as obtained from General Electric's description of the BWR/6 Nuclear Island. In this case, the piping and instrumentation diagrams contained data on pipe sizes, which are included in Tables II-3 and II-4.

Table II-1. Summary of PWR Reference Plant Piping Inventory

Building or Area Interconnected	Number of Lines
Annulus Building with --	
Atmosphere	4
Balance of Plant (BOP) Water	1
Containment Structure	58*
Control Building	3
Diesel Generator Building	2
Fuel Building	6**
Radioactive Solid Waste and Decontamination Building	48***
Turbine Building	19
Ultimate Heat Sink	15
Yard	8
Containment Structure with --	
Atmosphere	1 (air exhaust)
Turbine Building	2
Control Building with --	
Atmosphere	18 (9 ea. air intake and exhaust ducts)
Balance of Plant (BOP) Water	1
Normal Switchgear Building	1

* Includes four lines for Main Steam Line interfaces, four lines for Feedwater System interfaces, and four lines for Steam Generator Blowdown System interfaces shown in Table 7 (Ref. 1); also includes 2 air exhaust ducts.

** Includes two air intake and two air exhaust ducts.

*** Includes one air exhaust duct.

Table II-1. (Concluded)

Building or Area Interconnected	Number of Lines
Diesel Generator Building with -- Balance of Plant (BOP) Water	1
Fuel Building with -- Balance of Plant (BOP) Water	1
Radioactive Solid Waste and Decontamination Building	5
Yard	1
Radioactive Gaseous Waste Building with -- Turbine Building	2
Radioactive Solid Waste and Decontamination Building with -- Atmosphere	1 (air intake duct)
Balance of Plant (BOP) Water	1
Yard	2
Turbine Building with -- Atmosphere	18 (4 air intake and 14 air exhaust ducts)
Auxiliary Boiler Building	3
Balance of Plant (BOP) Water	1
Service Building	4
Yard	4

Table II-2. PWR Reference Plant Piping Inventory by System

System	Building or Area Interconnected	Number of Lines
Aerated Vent and Drain	Annulus Building with --	
	Containment Structure	1
	Radioactive Solid Waste and Decontamination Building	1
	Turbine Building	1
Annulus Building HVAC*	Annulus Building with --	
	Atmosphere	2 (1 ea. air intake ducts and exhaust ducts)
	Containment Structure	1 (air exhaust duct)
	Radioactive Solid Waste and Decontamination Building	1 (air exhaust duct)
Auxiliary Feedwater System	Annulus Building with Containment Structure	4
Auxiliary Steam and Condensate	Turbine Building with --	
	Annulus Building	3
	Auxiliary Boiler Building	3
	Radioactive Gaseous Waste Building	1
	Service Building	4
Boron Recovery	Annulus Building with --	
	Fuel Building	1
	Radioactive Solid Waste and Decontamination Building	1
	Yard	2
Carbon Dioxide Fire Protection	Annulus Building with --	
	Control Building	1
	Yard	1

*Heating, Ventilation and Cooling

Table II-2. (Continued)

System	Building or Area Interconnected	Number of Lines
	Control Building with Normal Switchgear Building	1
Containment Atmosphere Air-borne Radiation Monitoring	Annulus Building with Containment Structure	2
Containment Purge Air	Annulus Building with Containment Structure	1 (air exhaust duct)
	Containment Structure to Atmosphere	1 (air exhaust duct)
Condensate	Turbine Building with Yard	4
Condenser Evacuation	Turbine Building with - Atmosphere	2
	Radioactive Gaseous Waste Building	1
Control Building HVAC Cable Spreading Room Ventilation	Control Building to Atmosphere	4 (2 ea. air intake ducts and air exhaust ducts)
Control Room Air Conditioning	Control Room Atmosphere	2 (1 ea. air intake duct and air exhaust duct)
	Remote Room to Atmosphere	2 (1 ea. air intake duct and air exhaust duct)
Refrigeration Equipment Room Air Conditioning	Control Building to Atmosphere	2 (1 ea. air intake duct and air exhaust duct)

Table II-2. (Continued)

System	Building or Area Interconnected	Number of Lines
Diesel Generator Room Ventilation	Control Building to Atmosphere	2 (1 ea. air intake duct and air exhaust duct)
Electrical Tunnel Ventilation	Control Building to Atmosphere	2 (1 ea. air intake duct and air exhaust duct)
Emergency Switchgear Air Conditioning	Control Building to Atmosphere	4 (2 ea. air intake ducts and air exhaust ducts)
AII-7 Feedwater **	Demineralized Water Makeup	
	Annulus Building with - Fuel Building	1
	Turbine Building	1
	Yard with - Fuel Building	1
	Turbine Building	1
Fuel Building HVAC	Annulus Building with - Containment Structure	4 **
	Turbine Building	4
	Annulus Building with - Fuel Building	4 (2 ea. air intake ducts and air exhaust ducts)
	Atmosphere	2 (1 ea. air intake duct and air exhaust duct)
Fuel Pool Purification	Annulus Building with Containment Structure	2
Gaseous Vent and Drain	Annulus Building with Containment Structure	2
Instrument and Service Air	Annulus Building with - Containment Structure	4
	Turbine Building	2
	Containment Structure with Turbine Building	2

** Also Accounted for in Table 7 of Ref. 1

Table II-2. (Continued)

System	Building or Area Interconnected	Number of Lines
Main Steam**	Annulus Building with -- Containment Structure Turbine Building	4** 4
Primary Grade Water	Annulus Building with -- Containment Structure Radioactive Solid Waste and Decontamination Building Yard	1 10 4
Radioactive Liquid Waste	Annulus Building with Radioactive Solid Waste and Decontamination Building	10
Radioactive Solid Waste	Radioactive Solid Waste and Decontamination Building with -- Annulus Building Fuel Building Yard	20 5 2
Reactor Plant Component Cooling Water	Annulus Building with Containment Structure	16
Reactor Plant Gas Sampling	Annulus Building with -- Containment Structure Ultimate Heat Sink	5 13
Reactor Plant Gas Supply	Annulus Building with -- Containment Structure Radioactive Solid Waste and Decontamination Building	2 3

AII-8

** Also accounted for in Table 7 of Ref. 1.

Table II-2. (Concluded)

System	Building or Area Interconnected	Number of Lines
Reactor Plant Service Water	Annulus Building with --	
	Control Building	2
	Diesel Generator Building	2
	Ultimate Heat Sink	2
Reactor Plant Ventilation	Annulus Building with --	
	Containment Structure	1
	Radioactive Solid Waste and Decontamination Building	1
Radioactive Solid Waste and Decontamination HVAC	Radioactive Solid Waste and Decontamination Building to Atmosphere	1 (air intake duct)
Steam Generator Blowdown	Annulus Building with --	
	Containment Structure	4**
	Turbine Building	4
	Containment Structure with Radioactive Solid Waste and Decontamination Building	1
Turbine Building HVAC	Turbine Building to Atmosphere	16 (4 air intake ducts and 12 air exhaust ducts)
Water Fire Protection	Balance of Plant (BOP) Water to --	
	Annulus Building	1
	Control Building	1
	Diesel Generator Building	1
	Fuel Building	1
	Radioactive Solid Waste and Decontamination Building	1
	Turbine Building	1
Chilled Water	Annulus Building with Containment Structure	2

** Also accounted for in Table 7 of Ref. 1.

Table II-3. Summary of BWR/6 Reference Plant Piping Inventory

Building or Area Interconnected	Number of Lines	Size (in)
Auxiliary Building with - Atmosphere	7 (3 air intake ducts and 4 air exhaust ducts)	
Balance of Plant (BOP)	2	3
	4	4
	2	6
	2	10
	1	18
	2	24
Containment Structure	4	1
	4	1 1/2
	9	3
	5	4
	3	6
	1	8
	1	18
	1 air exhaust duct	--
Control Building	1	1
	2	2
	4	3
	4	6
Diesel Generator Building	2	1
	1	2
Fuel Building	5	3/4
	5	1
	2	1 1/2
	2	2
	8	2 1/2
	1	3
	5	4

Table II-3. (Continued)

Building or Area Interconnected	Number of Lines	Size (in)
Auxiliary Building with - (cont'd)		
Fuel Building (cont'd)	2	6
	5	10
	2	14
	1 personnel airlock	--
	1 air exhaust duct	--
Radwaste Building	2	6
	13	3
Turbine Building	1	4
Containment Structure with -		
Balance of Plant (BOP)	1	12
Fuel Building	8	3/4
	8	1
	5	2
	1	3
	1	4
	1	6
	1 air exhaust ducts	--
Control Building with -		
Atmosphere	2	2
	3 (2 air intake ducts	--
	and 1 air exhaust duct)	
Balance of Plant (BOP)	1	3
	2	4
	2	6
Div. 1 Diesel Generator Building	1	1 1/2

Table II-3. (Concluded)

Building or Area Interconnected	Number of Lines	Size (in)
Diesel Generator Building with - Atmosphere	2 (1 ea. air intake duct and 1 air exhaust duct)	-
Balance of Plant (BOP)	1	1 1/2
	1	2
	4	2 1/2
	2	4
	2	8
Drywell to Atmosphere	1 air exhaust duct	-
Fuel Building with - Atmosphere	2 air exhaust ducts	-
Balance of Plant (BOP)	2	2
	2	4
	4	6
	5	10
	1	12
	2	24
Containment Structure/Shield Bldg. Annulus	1 air intake duct	--
Diesel Generator Building	1	1
	2	1 1/2
	1	2
Radwaste Building	4	3
Radwaste Building with - Balance of Plant (BOP)	1	3
	6	6
Atmosphere	5 (2 air intake ducts and 3 air exhaust ducts)	.
Switchgear Building to Atmosphere	2 (1 ea. air intake duct and air exhaust duct)	.

Table II-4. BWR/6 Reference Plant Piping Inventory by System

System	Building or Area Interconnected	No. of Lines	Size (in)
Auxiliary Building Electrical Areas, Corridors, Steam Tunnel and Elevator Tower HVAC	Auxiliary Building to Atmosphere	5 (2 air intake ducts and 3 air exhaust ducts)	-
Auxiliary Building Emergency Core Cooling System Area Pressure Control	Auxiliary Building to Atmosphere	2 (1 ea. air intake duct and air exhaust duct)	-
Carbon Dioxide Fire Protection	Balance of Plant (BOP) CO ₂ Storage Tank with -		
	Div. 1 Diesel Generator Room	1	4
	Div. 2/3 Diesel Generator Room	1	4
Clean Radwaste Drain	Auxiliary Building with -		
	Containment	4	1
		3	1 1/2
	Radwaste Building	2	3
	Fuel Building with		
	Auxiliary Building	1	1
	Containment Building	8	3/4
		4	1
	Radwaste Building	1	3
Closed Cooling Water	Fuel Building with -		
	Containment Structure	1	2
	Radwaste Building	2	3
Control Building Chilled Water	Control Building with -		
	Atmosphere	2	2
	Auxiliary Building	4	6
		4	3

Table II-4. (Continued)

System	Building or Area Interconnected	No. of Lines	Size (in)
Control Building HVAC	Control Building to Atmosphere	3 (2 air intake ducts and 1 air exhaust duct)	-
Control Building Potable Water	Control Building to Balance of Plant (BOP) Water	1	3
Demineralized Water and Condensate Distribution	Auxiliary Building with -		
	Balance of Plant (BOP)		
	Condensate Storage Tank	1	4
		1	18
	Main Condenser	1	10
	Supply Header & Storage Pumps	1	4
		1	10
	Containment Structure	1	1 1/2
		7	3
		1	6
		1	8
		1	18
	Control Building	2	2
	Div. 1 Diesel Generator Battery Room	1	1
	Div. 2/3 Diesel Generator Building	1	1
		1	2
	Div. 4 Control Room Battery Area	1	1
	Fuel Building with -		
	Auxiliary Building	1	2
		1	4
		3	10
	Div. 1 Diesel Generator Building	1	2
	Containment Structure with Balance of Plant (BOP)	1	12
	Condensate Storage Tank		
	Containment Structure with Fuel Building	1	1
		2	2
		1	4
		1	6

Table II-4. (Continued)

System	Building or Area Interconnected	Nc. of Lines	Size (in)
Demineralized Water and Condensate Distribution	Radwaste Building with -		
	Balance of Plant (BOP) Main Condenser	1	6
	Balance of Plant (BOP) condensate storage tank/transfer pumps	1	6
Detergent Drain	Radwaste Building with -		
	Auxiliary Building	1	3
	Fuel Building	1	3
Diesel Generator and Switchgear Rooms HVAC	Div. 1, 2/3 Diesel Generator Buildings to Atmosphere	2 (1 ea. air intake duct and air exhaust duct)	-
	Div. 2/3 Switchgear Building to Atmosphere	2 (1 ea. air intake duct and air exhaust duct)	-
Diesel Fuel Transfer	Div. 1, 2 Diesel Generator Building with	1	1 1/2
	Balance of Plant (BOP) fuel oil transfer pumps	1	2
Dirty Radwaste Drain	Radwaste Building with -		
	Auxiliary Building	10	3
	Fuel Building	1	3
Essential Service Water	Balance of Plant (BOP) Cooling Pond Supply with-		
	Auxiliary Building	1	24
	Fuel/Auxiliary/Control Buildings	1	24/18/6
	Balance of Plant (BOP) Cooling Pond Return with-		
	Auxiliary Building	1	24
	Fuel/Auxiliary/Control Buildings	1	24/18/6
	Fuel Building with Div. 1 Diesel Generator Bldg.	1	1
		1	1 1/2

Table II-4. (Continued)

System	Building or Area Interconnected	No. of Lines	Size (in)
Fuel Building HVAC	Fuel Building with - Atmosphere	1 (air exhaust duct)	-
	Auxiliary Building	1 (personnel airlock)	-
Fuel Pool Cooling & Cleanup	Fuel Building with - Auxiliary Building	3	3/4
		4	1
		2	1 1/2
		2	10
		2	14
	Balance of Plant (BOP) Cooling Pond Supply	2	10
	Balance of Plant (BOP) Cooling Pond Return	2	10
	Balance of Plant (BOP) Main Condenser	1	10
		1	12
	Containment Structure	1	1
		2	2
		1	3
	Containment Structure with Balance of Plant (BOP) Cooling Pond Supply	2	2
Heated Water Distribution	Auxiliary Building with - Balance of Plant (BOP)	2	6
	Containment Structure	2	4
	Fuel Building	4	4
	Balance of Plant (BOP) Supply with Radwaste/ Auxiliary/Fuel Buildings	1	6/4/4
	Balance of Plant (BOP) Return with Radwaste/ Auxiliary/Fuel Buildings	1	6/4/4
High Pressure Core Spray Service Water	Balance of Plant (BOP) Supply Header with - Auxiliary Building	1	3
	Div. 3 Diesel Generator Building	1	8
	Balance of Plant (BOP) Return Header with - Auxiliary Building	1	3
	Div. 3 Diesel Generator Building	1	8

Table II-4. (Continued)

System	Building or Area Interconnected	No. of Lines	Size (in)
Hydrogen Mixing, Drywell Relief and Containment Vacuum Relief	Drywell to Atmosphere	1 (air exhaust duct)	-
Pneumatic Supply	Fuel Building with - Auxiliary Building Containment Structure	2 2	3/4 1
Radwaste Building HVAC	Radwaste Building to Atmosphere	5 (2 air intake ducts and 3 air exhaust ducts)	-
Reactor Island Non-Essential Chilled Water	Auxiliary Building with - Containment Structure	2 2 2	3 4 6
	Fuel Building	2	6
	Radwaste Building	2	6
Service Air Distribution	Auxiliary Building with Containment Structure	1	4
	Div. 2 Diesel Generator Building	1	2
	Fuel Building	1	2
		1	3
	Turbine Building	1	4
	Control Building with Div. 1 Diesel Generator Bldg.	1	1 1/2
	Fuel Building with Div. 1 Diesel Generator Bldg.	1	1 1/2
	Radwaste Building with Balance of Plant (BOP)	1	3

Table II-4. (Concluded)

System	Building or Area Interconnected	No. of Lines	Size (in)
Standby Gas Treatment (SGTS)	Fuel Building Div. 1 (SGTS) with -		
	Atmosphere	1 (air exhaust duct)	-
	Balance of Plant (BOP)	2	2
	Fuel Building Div. 1 & 2 (SGTS) with -		
	Auxiliary Building	1 (air exhaust duct)	-
Wet Standpipe Fire Protection	Containment Structure	1 (air exhaust duct)	-
	Containment Structure/Shield Building Annulus	1 (air exhaust duct)	-
	Balance of Plant (BOP) supply with -		
	Control Building	2	4
	Div. 1 Diesel Generator Building	2	2 1/2
	Div. 2, 3 Diesel Generator Buildings	2	2 1/2
	Fuel Building	4	6
	Radwaste Building	2	6
	Fuel Building with Auxiliary Building	8	2 1/2

APPENDIX III
ACCIDENT CONSEQUENCE ANALYSIS CODES

Although a great many computer codes have been developed for the purpose of reactor safety analysis, almost without exception these codes are applicable only to the design-basis and/or lesser accidents. The proliferation of these codes is of course guaranteed by the requirement for extensive safety analysis and documentation in the normal licensing process, as specified in 10 CFR 50. 33-34.

III. 1 REACTOR ACCIDENT CODES

Even with the aid of computer codes, the prediction of the performance of engineered safety features under accident conditions is difficult, and subject to substantial uncertainties, as attested to by the controversy surrounding the effectiveness of the ECCS. The basic weaknesses are two-fold; the shortcomings of the experimental data base, and the complexity of the phenomena which are to be modeled in the codes. In view of the difficulty of modeling the DBA numerically, it is not surprising that there is no single, simple, straightforward code, or other method, for the analysis of a Class-9 accident. Following the meltdown of the core, not even the basic geometry of the situation can be specified with confidence, and estimates of heat transfer rates and other interactions of the core mass with its surroundings become very uncertain.

III. 1. 1 Core Meltdown Analysis Methods

Two codes, BOIL and CORRAL, (with extensive hand calculation for extended times) were used by the Reactor Safety Study team in their analysis of Class-9 accidents, and are available to the present study. Basically the BOIL code estimates the time required for core meltdown after loss of coolant, and times of release of fission products to the containment atmosphere. CORRAL estimates the containment atmosphere's fission-product inventory as a function of time after accident initiation. However, BOIL does not perform a complete calculation of the meltdown process; only the initial phase is modeled in detail from first principles. After core slumping begins, recourse is made to hand calculations, based on engineering judgment and various plausible assumptions, in order

to obtain (a) estimates of the time at which the core slumps into the lower head of the pressure vessel, (b) the time to failure or meltthrough of the lower head, (c) the rate of penetration of the concrete foundation mat, etc. After the initiation of core slumping, core geometry (and therefore heat transfer rates) becomes very uncertain. Subsequent interaction with the pressure vessel, core support plate, possible water, and eventually the concrete foundation, makes the specification of physical properties, heat transfer rates, and penetration rates rather speculative. Hence, the timing of specific events such as meltthrough should be regarded as order-of-magnitude estimates rather than accurate predictions corresponding to a given accident scenario. In order to estimate heat transfer rates, for instance, one must start by assuming some simple geometric configurations for the core mass. References 1 and 2 provide discussions of the difficulties and uncertainties inherent in these estimates.

All of these uncertainties and unknowns are of interest here only to the extent that they bear on the principal consideration, namely, the release of radioactivity to the external environment. The CORRAL code provides estimates of the fission product inventory within the containment, potentially available for release upon breach of containment at any time. Basically, this code performs the required bookkeeping to record release and removal rates of fission product species within the containment. To do this, CORRAL requires a mass of input data from BOIL, and the associated hand calculations, including the time scale of the accident (times of critical events such as pressure vessel meltthrough, etc.), pressure-time and temperature histories of the containment atmosphere, times and rates of release of fission products, etc., all of which are subject to the substantial uncertainties just mentioned. The output of CORRAL is a specification of rates of release of radioactivity from the containment. In the case of a surface-sited plant, as in Reference 1, this release would be to the external atmosphere. For the present study, this release would be to the soil/rock exterior to the containment, then

(possibly) to the atmosphere. The estimation of migration or transport rates of radionuclides through soil or rock is fraught with possible greater uncertainties than those inherent in the input to (and hence output from) the CORRAL code. It is therefore necessary to consider carefully the applicability of the use of such codes in the present study.

III.2 GROUND TRANSPORT CODES

Of the several ground transport codes available, two which exemplify features considered to be appropriate to this study are TWIG and SOGMOD (Refs. 3 and 4). The TWIG code may be used for gaseous radionuclide dispersion analysis in multilayered media. The SOGMOD code may be used for ionic radionuclide dispersion analyses in multilayer media.

Ground transportation of radionuclide species can be effectively analyzed by hand computations using the mathematics of diffusion theory with a pressure term. The hand analysis can become cumbersome when large numbers of radionuclides are transported and when the soil properties are discontinuous. In these latter cases, the codes discussed above may be employed.

The geosphere transport model (Ref. 5) describes the transport of radionuclides by flowing groundwater following a leach incident at an underground nuclear waste disposal site. This model assumes that at some arbitrary time after the waste was deposited, the contents of the site are dissolved at a specified constant rate into an underground water stream. The water stream flows at constant velocity through a homogeneous one-dimensional column of the geologic medium and discharges to a surface water body. The dissolved radionuclides are assumed to be in absorption equilibrium at all points in the geologic medium. Radioactive decay (including chain decay of the actinides) is modeled both at the disposal site and during

migration through the geologic medium. Trace concentrations of the dissolved nuclides are assumed, and as a result the absorption equilibrium constants are independent of concentration. A constant axial dispersion coefficient is also assumed. Molecular dispersion in still water is not treated. This model and the Sandia ground transport code (Ref. 6) are similar.

The TWIG code calculates two-dimensional ideal gas flow in porous media. The media need be neither homogeneous nor isotropic. Pressure-time histories as well as effects of gaseous condensation are also included.

SOGMOD calculates ionic radionuclide concentration time histories in porous media. The medium need not be homogeneous nor isotropic. Hence, the effects of channelling on ionic radionuclide migration could be evaluated provided the ionic radionuclide channel transport properties could be established. Both absorption and radioactive decay are modeled as well as the effects of a gravity pressure head and moisture content.

III. 3 ATMOSPHERIC TRANSPORT CODES

Of the atmospheric transport codes discussed below, there is no clear advantage in selecting one for use over another. In addition, there may be many additional meteorological codes which would be appropriate for this analysis. However, all of the codes suffer from the drawback of not being able to predict the weather. Even when actual areal-time weather data are used, the code predictions agree rather poorly with actual measurements of distributions of contaminations.

The Atmospheric Diffusion Particle-in-Cell (ADPIC) code (Ref. 7) is a numerical, three-dimensional, Cartesian, particle-diffusion code capable of calculating the time-dependent distribution of air pollutants under many conditions. These conditions include strongly distorted advection wind fields, calm conditions, space-variable surface roughness, wet and dry deposition, radioactive decay, and space- and time-variable diffusion parameters.

Basically, the code solves the three-dimensional advection-diffusion equation in its flux conservative form (pseudovelocitv technique) for a given non-divergent advection field by finite difference approximations in Cartesian coordinates. The method is based on the particle-in-cell technique with the pollutant concentration represented by Lagrangian-marker particles inside a fixed Eulerian grid (Refs. 8, 9, 10, 11).

Most air pollution scenarios involve time- and space-varying advection fields (shear) and diffusion parameters. They involve topography, precipitation, and gravitational effects for a variety of active or inert sources and are inherently three-dimensional in nature. ADPIC was developed to model these aspects of pollutant dispersal over short or long time spans for any specified source term with the exception, for the present, of photochemistry.

With the development of the three-dimensional non-divergent (mass-conservative) windfield model MATHEW (Ref. 12), which is used to provide the full three-dimensional space- and time-varying advection field to ADPIC, pollutant dispersion studies of any desired complexity can be undertaken.

ADPIC has undergone extensive validation tests against closed analytic solutions and regional tracer studies (Lange, 1973) and compares to within a 5% error against selected analytic solutions. Agreement is remarkably consistent against methyl iodine tracer studies at NRTS, Idaho Falls, Idaho, and Ar-41 plumes at Savannah River Laboratory, Aiken, South Carolina. ADPIC concentrations are within a factor of 2 about 60% of the time, and within an order of magnitude of measurements 95% of the time without any tuning of the model to any specific site, tracer scenario, or tracer material. These results, although still only partially validated, provide an indication of the degree of accuracy with which the ADPIC-MATHEW package can compute complex regional pollutant concentration and deposition distributions.

The DEfense Land Fallout Interpretive Code (DELFI_C) is a sophisticated modularized computer code (Ref. 9), designed to predict fallout for nuclear weapon burst yields ranging over many orders

of magnitude with several types and percentages of fission and for all heights and depths of burst. DELFIC estimates the soil burden lifted by the burst and the size distribution of fallout particles as well as the cloud rise and final cloud configuration. It then computes the transport and settling of the fallout particles under the effect of input space and time variant wind fields and decay chains. DELFIC can yield any of a number of desired fallout indices such as mass deposited, time of onset, normalized or unnormalized exposure rate at any time specified, activity contributed by individual mass chains, exposure accumulated over specified time limits, etc.

DELFIC requires atmospheric data to be provided at arbitrarily selected z-direction levels over ground zero in order to compute the cloud rise and configuration. The atmospheric data required consist of temperature, pressure or air density, and relative humidity. Wind speed and direction data used to compute the particle transport are input on an arbitrarily selected x-y grid with the ground zero location being specified by its x-y coordinates. DELFIC will accept wind data in vector or component form. If the wind data are in component form, different wind fields can be specified for any or all points of the grid; if the wind data are in vector form, the winds at ground zero are assumed constant throughout the user defined x-y grid. Wind data can also be supplied as applicable for a specified time period only, after which new wind data may be input for the next desired time period. The z-direction levels must begin at or below the height of burst level and should continue to above the expected cloud height. DELFIC linearly interpolates all non-wind atmosphere variables above the input values using the ARDC standard atmosphere to obtain input distributions up to a height of 50,000 m.

In addition to atmospheric data, DELFIC requires information as to the height of burst, soil type, fission type, the number of particle size classes, particle density, etc.

AIRDOS is a FORTRAN IV computer code (Ref. 4) used to estimate annual population doses (man-rem) and the maximum annual individual doses in an area as a result of releases of radionuclides from a nuclear facility. The code can estimate doses resulting from simultaneous releases of as many as 36 radionuclides from as many as six plant stacks or roof vents. Pathways of exposure to man included in AIRDOS are (a) inhalation of radionuclides in air, (b) immersion in air containing radionuclides, (c) exposure to ground surfaces contaminated by deposited radionuclides, (d) ingestion of food produced in the area, and (e) immersion in contaminated water. Doses are estimated for 11 organs, including the total body.

The area surrounding the nuclear facility may be described as a 20 x 20 square grid with the facility at the center. The grid size is selectable, human population, numbers of beef cattle and dairy cattle, and identification as to whether the zone is a farming or a water area is required as input data for each of the 400 grid squares.

Meteorological data are supplied as input for the atmospheric dispersion model used to estimate concentrations of the radionuclides in air and the rates of deposition on ground or water surfaces. A terrestrial model developed by Booth and Kay is used in AIRDOS to estimate ingestion of radionuclides through food chains to man. Ingestion of fish or other food produced from water areas is not included in the present version of AIRDOS.

External doses from gamma radiation from overhead plumes are not estimated. This additional external exposure may be important in the immediate vicinity of a nuclear facility, especially under stable atmospheric conditions. Therefore, it should be estimated separately where conditions warrant and added to the external doses estimated by AIRDOS.

Population doses are summarized in the output tables of AIRDOS in a number of ways - by nuclides, pathways, and organs. If more than 36 radionuclides must be considered in a source term, two or more computer

runs can be made and the data in the summary tables from each run simply added together. The highest individual doses in the area for each organ are tabulated for each radionuclide and the highest organ doses from all radionuclides in the source term are listed. The location of the highest individual dose is specified.

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