2340 Richards Avenue
Idaho Falls, Idaho 83401
September 21, 1974



Ms. Dixie Lee Ray
USAEC
1717 H Street NW
Washington, D.C. 20545

Dear Ms. Ray:

I am resigning my position as an Associate Scientist with Aerojet Nuclear Company in order to be free to tell the American people the truth about the potentially dangerous condition in the nation's nuclear power plants. As an employee of Aerojet Nuclear I have not been able to freely express my concerns about the nuclear reactor safety issues. Consequently I will be working for the Union of Concerned Scientists in an attempt to more fully inform the public about the current state of knowledge concerning reactor safety, particularly the emergency core cooling systems.

I have been employed at the Idaho National Engineering Laboratory for the past seven years for Aerojet Nuclear and its predecessors. During that time I have been involved in the development of computer codes which are used in the thermal-hydraulic predictions of loss-of-coolant situations. I was the principal author of the THETA1-B code which was adopted by the AEC as an accepted method of predicting the thermal behavior of a fuel rod during a LOCA. The last several years I have been working on a new thermal-hydraulic loop code. The primary goal of this project is to develop analytical models which will more realistically describe the physical processes that could occur during a LOCA.

While analytical models for predicting the fluid behavior during a LOCA have been developed by both the nuclear industry and the AEC the techniques in general are not capable of describing actual physical situations with a reasonable degree of reliability. The AEC is using shaky and unproven computer predictions as a basis for answering such vital questions as the effectiveness of reactor safety systems in preventing catastrophic accidents. This is wholly unacceptable.

Adequate experimental programs to determine the workability of reactor safety systems are also urgently needed. Experimental verification of the analytical computer codes is a necessity if we are to place our faith in these methods.

Aerojet Nuclear employees were used by the AEC as consultants during the ECCS hearings. In 1971 the AEC adopted the methods we had developed, but completely ignored our reports concerning the serious limitations of those methods. They were the best that could be develope based on the limited analytical and experimental research the AEC and nuclear industry had carried out, but they were preliminary and definite not an adequately proven way of determining nuclear reactor safety. Little has changed in the past few years, and the safety of nuclear reactors is still uncertain and unverified.

The AEC is ignoring advice from many of its experts on reactor safety problems, a situation that has given rise to numerous resignations. Several of my colleagues have gone to work trying to help the

To GM for preparation of
reply signature block omitted

Copy of incoming to Chairman Ray, Mr. Fraley
and Dr. Hanauer

Cy ... to Mr. Kouts & Mr. Foucharo

utility companies understand the reactor safety problems that the AEC would prefer to ignore, but I believe that the general public, and not just the companies investing in nuclear generating equipment, must be told the truth about the potential hazards.

I also have personal reservations concerning the radioactive waste problems. While I am not an expert in waste management I find the long term radioactive waste question deeply disturbing. The present generations get the electricity from nuclear plants and we leave the radioactive wastes for our children and future generations to take care of. Plutonium, an extremely hazardous material that retains its radioactive potency for hundreds of thousands of years, is hardly a legacy that future generations should be given.

In spite of the soothing reassurances that the AEC gives to an uninformed, mislead public, unresolved questions about nuclear power plant safety are so grave that the US should consider a complete halt in nuclear power plant construction while we see if these serious questions can, somehow, be resolved. The most prudent course of action that we can take is to proceed cautiously.

Sincerely,

Carl J. Hocevar

Carl J. Hocevar

cc: Dan Ford, Union of Concerned Scientists

MINUTES OF
SAFETY FEATURES PROVIDED BY ARCHITECT-ENGINEER
(GILBERT ASSOCIATES INC.) SUBCOMMITTEE
WASHINGTON, D. C.
AUGUST 23, 1974

The ACRS Subcommittee for Safety Features provided by the Architect-Engineers was held at 1717 H Street, N.W., Washington, D.C., was convened at 2:00 p.m., Friday, August 23, 1974. Dr. Isbin was chairman. Dr. Lawroski and Mr. Bender were also present.

1.0 MEETING WITH REPRESENTATIVES OF GILBERT ASSOCIATES INC. (meeting open to the public)

1.1 Chairman's Opening Remarks

Dr. Isbin called the meeting to order and informed the attendees of the purpose of the meeting and the rules under which the meeting was being conducted. He noted that Gary Quittschreiber was the Designated Federal Employee and that a notice of the meeting was published in the Federal Register on August 7, 1974. He indicated that no requests for oral statements based on previously written statements had been received.

1.2 Role of the Architect-Engineer

Mr. Hans Lorenz, Vice President and General Manager of the Utilities Division, described Gilbert Associates Inc. (GAI) role as follows:

- GAI has about 2,400 people in Reading, Pennsylvania and about 1,400 people in Jackson, Michigan. The total organization at the present time is around 3,800 people. This includes professional and non-professional people. Out of this 3,800, only about 1,900 are involved in the design of nuclear and fossil fuel power plants. About 90% of the engineering for power plants is done in the Utilities Division and the Energy Conversion Division. Attachment 3 shows a breakdown of all of the divisions.

- Attachment 4 shows the organization of the Utilities Division. In addition to the Staff groups, there are essentially five large groups of engineering departments and projects. Each one is headed up by an engineering manager. Each engineering manager heads up a group of engineering departments.

- Attachment 5 shows a typical chart of a project team organization. The project is headed up by the project manager who is the focal point responsible to the engineering manager of Gilbert Associates and responsible to the client. The project manager has assistance through project control engineers who are involved in scheduling, QA, estimating, and others.

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- GAI stresses the team effort. They try to select a team which is compatible, a team which has experience in nuclear work among the key people such as the project manager, senior project manager and project engineers.

- The Utilities Division has a total of 23 departments. See Attachment 6 for a breakdown of the Utilities Division. There are a total of about 750 people in the Engineering Departments. About 550 of the 750 are professionals, either with a degree or licensed engineers. About 19% of the 550 have advanced degrees, the rest of the personnel (about 386) are draftsmen.

- Attachment 7 shows GAI's growth in manpower. It shows that engineering has increased at a much larger pace than drafting. The reason for this is due to the amount of analysis required.

- Attachment 8 shows man-hours per month expended on three different nuclear projects. Each of the three successive projects shows a higher manpower workload. The last project shows a very pronounced early peak compared with the second one. Part of the reason for the early peak is the new format of the PSAR. The last project will take about one and a half million man-hours, whereas the first project required only a few hundred thousand man-hours. The reasons for the progressively larger curves include greater sophistication, better analytical tools, client participation which affects the man-hours and AEC requirements.

- Attachment 9 shows the GAI nuclear workload. The nuclear work is about 60% of GAI's power plant work, the balance is fossil work.

- GAI is fully competent and qualified to design systems and structures. They are not qualified to design components; they never want to design components although they may have the capability to analyze components and have a good feel of what it takes to design a component or what it takes to manufacture a component. They intend to stay out of the business of component design.

In response to a question from Mr. Bender concerning difference in workload between foreign and domestic projects, Lorenz said there was a large difference. He said even if you compare nuclear islands between domestic and foreign, you see less man-hours and less duration on the foreign unit. He felt this was due to the involvement of the foreign AEC which is much more passive and the questioning is more reduced. He didn't think they had to attend any foreign AEC hearings on these units except for one case.

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In response to a question from Dr. Isbin concerning a comparison of the degree of quality assurance and margins of safety on the foreign projects, Lorenz said the amount of documentation, the amount of paper work, the types of analysis, the engineering, the internal checks and the design control work are about the same, but the requirements for documentation are less.

## 1.3 Standardization

### 1.3.1 General Design and Pre-Designated Site Criteria

Mr. S. Goodman, Engineering Manager - Utilities Division, mentioned the following significant items concerning design work in general with regard to standardization:

- The first step in the formal implementation of GAI standardization methods took place when they established a design control procedure committee in March 1972. Design control procedures represent the administrative control of design. There are about 46 planned design control procedure of which 36 are issued.

- GAI has a topical report program with the Commission with about 25 Topical Reports planned. Only two are in process at the present time.

- In 1973, GAI formed a committee and after speaking with the AEC and visiting all of the NSSS suppliers, they came to the conclusion that they would start with Phase 1 of a two-phase program leading to the submittal of a BOP. Phase 1 is essentially to develop the criterion concepts, to define a detailed scope split between GAI and the NSSS suppliers standard package, and to prepare a plan for Phase 2.

- They have certain basic concerns which prevent them from accepting the responsibility of a complete BOPSAR at this time (Phase 2 concept). First, is client acceptance of their standard design, second, is the AEC Staff acceptance, third, is the cost, including AEC filing fees, and fourth, is options and alternatives (finish time etc.). The decision for Phase 2 has not been made yet but is anticipated for late fall of this year. It may be delayed in view of the market situation in the utility industry at this time since pre-designated sites go along with standardization.

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- GAI feels a necessary part of the standardization program is the concept to remove the environmental, site and safety issues from the plant licensing process by instituting a separate licensing effort in advance of plant selection. The objective with standardization is a six-year schedule on the plant site. They will still have preceded this with three years in advance of that for pre-designated sites.

In response to a question from Mr. Bender concerning specifying plant requirements for pre-designated sites, Goodman said the designated site would develop with envelope types of specifications. It would probably receive more than one concept with a different number of units.

1.3.2 Design Control and Assurance

Mr. R. Mathys, Manager of Design Control, noted the following significant items concerning how GAI maintains control and standardization over its design activities:

- Corporate quality assurance, of which design control is a facet, is a responsibility assigned by the president to the executive vice president, to whom the operating divisions report. These divisions include the Construction, Services Quality Assurance, and Utilities Divisions. Each is operated as a profit center by a general manager.

- Within the Utilities Division there are five engineering managers (see Attachment 11), who report to the general manager and vice president of the division, and to whom the engineering departments in turn report.

- Under the executive vice president's chairmanship, a corporate quality assurance policy committee exists consisting of the executive vice president and the managers of the Procurement, Quality Assurance, Construction Services, and Utilities Divisions. This committee meets monthly. Their purpose is to monitor and direct the corporate Quality Assurance plan, to assure co-ordination between divisions, and to develop and disseminate decisions on QA policy. This committee also implements audits of the Quality Assurance Division, which is, in turn, responsible for auditing the other divisions to assure their management of adequate implementation of the corporate QA plan.

- Engineering design guides provide a standardized method for the solution of engineering problems, including calculational techniques, and margins to be used. Department standards assure uniformity of detail design implementation. Guide specifications standardize portions of procurement specifications which are not uniquely dependent on the specifics of a particular power plant.

- Each project has its own project management manual which is approved by a Gilbert Associates project manager and an engineering manager as well as by the client. The mandatory content of this manual is established in the corporate QA manual and in subordinate procedures. It includes information on their scope of work, document distribution, scope of client review, and interface control between Gilbert Associates and other organizations.

In response to a question from Mr. Bender concerning audits of the design organizations, Mathys said the audit team would obtain copies of the organization's procedures and then prepare a check list. They would then compare the procedures for specific compliance to the requirements of either the design control procedures or with the procedures they have written to govern their affairs. Mathys said a complete audit of the design organization has not been done yet but the audits are on schedule. Shield added that audit findings may be in relation to department performance and not necessarily related to a nuclear project.

1.3.3 How the Standard Plant Evolved and Containment Concepts

Bob Hottenstein, Project Manager for the Standard Plant Project, noted the following concerning the evolution of the standard plant design:

- Began in late 1973 and early 1974 after the AEC announced its standard plant program.

- A survey showed no specific client interest. GAI felt it would be beneficial to them to provide some effort to standardization.

- GAI's overall schedule for the standard plant is shown on Attachment 12.

- Attachments 13-15 show the basic objectives of each of the five phases from development through final acceptance of the standard plant.

- Internally, both Commonwealth Associates and Gilbert Associates are participating in the Standard Plant Design. Whatever is

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originated by Commonwealth Associates will be reviewed by Gilbert Associates and whatever is originated by Gilbert Associates will be reviewed by Commonwealth Associates.

- o GAI is meeting with Utilities and NSS vendors to get their advice and guidance.
- o GAI cannot standardize to the degree of getting the two loop and four loop PWR systems with their safety trains into the same geometrical shield.
- o GAI is looking at both a cylinder and a sphere for the final containment envelope. Attachments 16-19 show the standard plant concepts for both the spherical and cylindrical containments. The spherical concept allows reduction in the size of the plant. Both concepts have a single mat for the reactor and auxiliary buildings.

1.4 Interfaces

The Subcommittee heard presentations on interfaces with the NSSS Supplier and was informed that:

- o The NSSS scope of supply can vary over a wide range depending, to a large degree, on the options the utility wants to pick up. Moreover, it depends to some extent as to when GAI is engaged to do the plant design.
- o In the past, the scope definition has been vague, but in the succeeding years, the interfaces are becoming more clearly defined at the outset of design. For those interfaces that need definition, the following procedure is followed for custom plants:

1. Identification - The undefined interfaces are identified by reviewing the NSSS-client contractual documents.
2. Discussion - The Client, NSSS and GAI conduct three-way discussions to review outstanding issue(s).
3. Scope Definition - The Client decides on the placement of each issue in either the NSSS or the BOP scope.
4. Design Assignment - Once a decision is reached, the interface is given to appropriate GAI Project Engineers who are then responsible for squaring away the technical parameters that relate to their individual disciplines.
5. Safety Recheck - Safety-related interfaces are further discussed and resolved before incorporation into the Client's SAR.

- The situation on the standard plant is different in that they have no contract with a client. Moreover, the NSSS scopes of supply vary because of supplier options. In the interest of minimizing the number of these options, GAI is defining those systems to be included in their standard plant. A client will still have the opportunity to select systems other than those they have placed in their BOP scope. If substitutions are made, the design direction will be clear because a point of departure will already have been defined.

- Attachments 20 and 21 show the system scope for the NSS and the BOP.

- Attachment 22 shows an example of the interface split for two systems.

In response to a question from Mr. Bender on what the GAI Quality Assurance Organization does about interfaces, Mathys stated that they have procedures that cover the requirement for checking of the criteria. He noted that Quality Assurance would not get directly involved in the technical interface but assures that it is handled properly and performs an audit function to see that this can be demonstrated.

1.5 Operating Experiences

The Subcommittee heard a presentation from Bill Meek on how operating experiences are factored into the design. He noted that:

- The usual sources of information, which are available to all Architect-Engineers are: (see Attachment 23)
  - Reactor Operating Experience Reports (ROE's)
  - Reactor Construction Experience Reports
  - Regulatory Operations Bulletins
  - Monthly Operating Reports from Client's Plants
  - Nuclear Power Experience
  - EEI Reports
  - Professional and Industry Conferences

- Other sources which are not public information include: (see Attachment 24)
  - Reports from Start-Up/Test Personnel
  - Continuing Service Projects
  - Utility OPerating Staff
  - Design Engineers Assigned to Operating Plants for Training

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- o Dissemination of information to the proper personnel is the responsibility of the respective Department Head. Reports are routed to department personnel and pertinent items are discussed in monthly department meetings. If appropriate, Engineering Design Guides or Guide Specifications are created or modified to assure avoidance of the problem on future designs.

- o For the case of serious failures, past designs are reviewed to determine if modifications are necessary. An example is the extensive modification of the main steam safety valve mountings at Ginna and Three Mile Island, which was a direct result of the failures experienced at Robinson and Turkey Point. All future designs of safety valve mounting are governed by an Engineering Design Guide, assuring that all forces and moments are properly considered in the design.

- o Although not a result of actual operating experience, is the orientation of the turbine-generator. On all future projects the turbine-generator shaft will be oriented radially from the reactor to minimize the risk from turbine missiles.

In response to a question from Mr. Bender concerning what action GAI took concerning leaking steam generator tubes, White indicated they do not get involved in steam generator chemistry but they are advising their clients to be able to remove the steam generators.

## 1.6 Research and Development

Bill Shields, Engineering Manager in the Utilities Division, informed the Subcommittee of the following concerning research and development:

- o GAI defines R&D as the application of corporate resources:
  1. to obtain basic data about the physical attributes and dynamic response of materials and working fluids
  2. to obtain new engineering correlations from existing data
  3. to design systems, structures, and components through the use of new or improved technical data or correlations
  4. to study the accomplishment of design by advanced methods and equipment for plant betterment and schedule improvement
  5. to apply systems, structures, and components to situations in which criteria are at significant variance with previous practice

- o In the Utilities Division, nuclear power plant development R&D activities are strongly oriented toward immediate design, construction, and plant operational problems; although, pure research activities are carried out where potential advantages of promising ideas are foreseen.

- Related to their R&D efforts and associated to their design activities are the development of computer programs. These programs are vital, in that they permit GAI to unify and control activities in-house, as opposed to subcontracting.

- About 1% of their direct client billings are invested in R&D on any given project, and about 5 or 6% of their total divisional manhour expenditure is devoted to this and to services supporting their in-house R&D.

- The A-E effort for a nuclear project represents only 2.5 to 4% of the overall client cost. Nuclear plant design features requiring R&D can be identified by any of the A-E's active in this field. In view of this, GAI believes it would be best once these needs are identified, for nuclear power plant applicants through groups such as EPRI to jointly fund such programs. The resulting reduction in overdesign would thus be of benefit to the industry as a whole rather than to any particular A-E. GAI believes their participation should be limited to the monitoring and identification of such programs.

- GAI feels the following R&D items should be considered for industry programs:

  1. A program to determine the effect of certain demineralizers in removing specific isotopes.
  2. Radwaste evaporator development for better methods of handling concentrates.
  3. Two-phase multicomponent flow and discharge correlation studies.
  4. Distribution and flow of gases in stagnant and turbulent atmospheres.
  5. Investigate in-plane sheer capacity for concrete.
  6. Confirm ballistics penetration formulae for nuclear plant applications.
  7. Basic experimental and analytical research related to missile target interaction with emphasis on the quantification of the dissipation of missile kinetic energy at impact into three parts: missile deformation, penetration, and structural deformation of the target.
  8. Develop materials characteristics of higher strength steels for containment applications.
  9. Resolution of $P_1$ probability for turbine missiles.
  10. Generating amplified (floor) response spectra from a ground spectrum.
  11. Investigate concrete material characteristics under prolonged exposure at temperatures greater than 200°F.

- GAI has no planned approach for evaluating the ECCS System.

- GAI is doing analytical work for the Germans on the ability of the compartments to hold pressure. The Germans are running the actual tests.

In response to a question from Mr. Bender concerning seismicity studies in the eastern states, Cronebergar stated that they feel a comprehensive study is needed but GAI as an A-E is not in a position to perform the study.

2.0 INDUSTRIAL SECURITY (closed session)

Mr. Raquet, GAI Security Officer, discussed designs to improve industrial security and informed the Subcommittee of the following:

- Industrial security considerations for protection of nuclear plants became a matter for concern five or six years ago, when it became evident that there are people who will use any means, including destructive measures, to gain their personal or political goals.

- GAI became concerned about the possibility of sabotage of the equipment needed for the safe operation of a nuclear plant. Procedures were then developed to help protect the safety equipment from sabotage.

- Regulatory Guide 1.17 and ANSI Standard N18.17 spell out the design features and operating procedures that make an effective security system.

- GAI believes that the ANSI Standard is in substance a necessary and valuable document. The problem with the standard is the lack of specific requirements for design. The problem is amplified by the fact that nuclear plant security designs are proprietary, and are not available for widespread study. Therefore, it is difficult to establish the so-called design state of the art.

- GAI feels that the development and adherence to proper written procedures for the operation of the security system is as important or perhaps more important than any feature they could design into the plant.

- GAI determines what areas are vital areas as defined by the ANSI standard. Such areas are the containment vessel, the fuel handling area, the control room and emergency water intake. They then must determine the location of the security fence. It should surround all vital areas. In some cases, the substation and cooling towers are outside the security fence. See Attachment 25.

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- Intrusion detection, lighting and television surveillance make up the first line of defense. Both intrusion detection and surveillance television are recommended. Television alone as an intrusion detection device is not effective.

- Consideration must be given to exterior construction and openings. It is common practice today for exterior doors to be of heavy metal with no opening hardware on the outside. Selected entrances, must have opening devices or security guards operating them. All exterior doors should be alarmed. Card key devices are used to supervise unguarded doors where access is desired and can be used for access control within specific areas of the plant.

- In the spherical containment, GAI is considering putting the safety equipment within the containment structure. See Attachment 26.

- One of the primary threats, as defined in ANSI N18.17, is a disgruntled employee who already has access to the plant. GAI proposes both greater access control getting into the plant as well as within the plant. One method of control is dual access. The inner control point would have to be opened by someone already in the plant. This would preclude overpowering a guard and gaining access Other control features would be internal access points where access between areas could be controlled. See Attachment 26A.

- GAI believes the most important security considerations are the type of construction, the plant layout and the design for access control.

## 3.0 <u>CONTINUATION OF MEETING WITH REPRESENTATIVES OF GILBERT ASSOCIATES INC. (meeting open to the public)</u>

### 3.1 <u>Regulatory Guides</u>

Lynn Myers, Licensing Engineer in the Environmental/Regulatory Department of Gilbert Associates, informed the Subcommittee that:

- Gilbert feels that the Regulatory Guides are a useful tool in designing a nuclear power plant. They consider it helpful to know what assumptions the staff will make in analyzing an accident. This allows them to predict the Staff's results with more accuracy and gives them an earlier opportunity to modify a design if required. Likewise, if there are several design options available, knowing what the Staff considers unacceptable allows them either to avoid that option, or to build a better case to support it.

- GAI believes, in some cases, the Staff is using the Regulatory Guide series as a short-cut method of implementing new "pseudo-regulations". They understand that the guides do not have the force of a regulation and that the Staff claims to be willing to look at alternatives. They have seen little, if any, tendency toward flexibility on the part of the Staff. What they do see is a tendency on the part of the Staff to say that if they do not commit to comply with a Regulatory Guide, the Safety Evaluation Report will be delayed.

- Another difficulty with the Regulatory Guides is their use by the Staff to push the state of the art into uncharted areas.

- In today's market, where the suppliers have more orders than material, the suppliers have no incentive to attempt to comply with requirements that carry them into areas which are new to them.

- Applicants are being asked to commit to issued guides even though the industry may at the time be commenting on them.

- A problem area has been their uncertainty about how late in a job a new Regulatory Guide must be incorporated in the design to avoid lengthy discussions during the Operating License review. It is estimated about 40% of the engineering on a job is completed at the time the Construction Permit is issued. Incorporating a new guide at this point could cause substantial rework. The new section on Implementation, found in some Regulatory Guides, can alleviate this concern if properly applied.

- GAI has developed a procedure, which is undergoing management review, to develop a Utilities Division position on each guide. If the position developed is felt to deviate from the guide, Gilbert will request a meeting with the Staff to determine the acceptability of the position to the Staff.

3.2 Turbine Generated Missiles

George Kowal, Manager of the Nuclear and Safety Analysis Department and Don Croneberger, Chief Structural Engineer, informed the Subcommittee of the following:

- It is generally assumed that turbine missiles originating from the end stage turbine wheel failures are emitted uniformly within 25° of the plane of rotation of the failed wheel. Missiles generated due to inner disc failure are more constrained with regard to emission angles and are postulated to occur uniformly within ± 5° of the wheel rotational plane. See Attachments 26B and 27.

- For a potential safety component or "target" to suffer a direct hit, the "target" must be located within the emission sector formed by the two outermost low pressure end stages. Any target that is located so that the path of the missile must first perforate the turbine pedestal is excluded from consideration; i.e., the reinforced concrete is assumed to provide adequate shielding.

- The "lob shot" missile trajectory exists for all plant locations. Previous studies have indicated that "lob shot" probabilities for site targets are typically two orders of magnitude less than those of "direct shots." When the "lob shot" strike probabilities are combined with the probability of turbine missile generation, it is found that relative to direct shot contribution for a particular target, no significant safety hazards exist.

- There is still uncertainty in the minds of many in the industry about the appropriate magnitude to use for $P_1$. The Gilbert position at this time is to use the vendors' recommendation and to analyze the plant site and arrangement for the turbine missile problem rather than attempt to design all plant structures, which implies "hardening", for the highly energetic destructive overspeed missile. Using the vendors' values for $P_1$, it is somewhat academic to conclude that the overall probability, $P_4$, is acceptably low for a plant layout with a "tangential" orientation. See Attachment 28.

- For domestic plants which are well along in the licensing process with "tangential" orientations, they are performing studies to determine the order of magnitude of possible $P_4$ values.

- If the eventual conclusion of the AEC Staff and the ACRS is that the turbine vendors' values for $P_1$ are somewhat low, Gilbert Associates agrees that the "radial" orientation illustrated in Attachment 29 should be used to reduce overall damage potential. It was noted that this orientation reduces the probability by approximately one to two orders of magnitude, which is a small difference when considering the uncertainties in the probability terms. As an alternate, a local shield on the turbine pedestal might be utilized with the "tangential" orientation to reduce damage potential.

- GAI does not consider "hardening" all plant structures to be an acceptable or practical solution.

- The problem of quantifying $P_3$ is quite complex. The precise value of $P_3$ depends on the degree of protection afforded a safety system by structures that would first have to be impacted, perforated, or otherwise breached or collapsed before safety systems on a particular trajectory might be endangered. See Attachments 30 and 31.

- The data presented in Attachments 32 and 33 illustrate the perforation thickness of concrete required for a selected set of tornado and destructive overspeed turbine missiles, the perforation thickness being, by definition, the thickness required to reduce the missile velocity to zero. The drastic difference of required thickness is self evident and the 4 to 5-foot thicknesses required for typical destructive overspeed turbine missiles implies the impracticality of "hardening" much of the plant site with individual concrete barriers (the cable values do not reflect any consideration for overall strength and stability of structures and associated reinforcing steel requirements).

- In contrast to analyzing or "hardening" structures all over the plant site to achieve acceptable $P_4$ values, Gilbert has investigated, to a limited extent, a concept of a reinforced concrete shield located on the edge of the turbine generator pedestal (See Attachment 34). This concept is applicable to the "tangential" orientation, and is not intended to provide protection for "lob shots". The extent of the wall in plan length and vertical height would depend upon the "direct shot" shielding required for a given plant layout. The intent of the design would be to provide partial protection for the "direct shot". The shield would be detrimental to normal plant operation and maintenance and the effect would only improve the overall probability by approximately one order of magnitude.

3.3 Qualification of Equipment to Operate In A Post-LOCA Environment

Thomas McMahon, Chief Electrical Engineer in the Utilities Division, discussed the following:

- GAI does not itself become involved in actual qualification testing for post-LOCA conditions, they do work closely with equipment suppliers to ensure that the design criteria is understood and correctly applied.

- GAI recieves assurance from each supplier that the particular piece of equipment has been qualified to perform in post-LOCA conditions, they must analyze the interconnected system to determine the adequacy of the equipment group to perform.

- GAI must check material compatibility, especially in those plants in which a potential exists for gas generation due to corrosive spray solutions. They must also locate equipment away from areas of physical damage should a LOCA occur or a high pressure line be broken.

- An A-E often has more flexibility than the NSSS on the selection of qualified equipment for his scope of supply since he has no commercial tie-in with a particular vendor.

- GAI's involvement with industry standards groups and regulatory agencies helps them establish and monitor the performance requirements for equipment vendors.

3.4 Radwaste Management with Respect to System Design and Design Criteria

Don White, Supervising Mechanical/Nuclear Engineer, informed the Subcommittee that:

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- GAI has the capability of supplying complete radioactive Waste Treatment Systems for liquid, solid and gaseous wastes. This includes design, engineering, procurement, start-up and test of the systems to meet the operational and performance objectives of the systems.

- The NSSS vendor provides the primary coolant concentrations and steam activities based on one percent failed fuel for PWR's and 100,000 microcuries per second after 30 minutes decay for BWR's.

- Attachment 35 shows a typical process flow diagram for a Liquid Radwaste System. This example is for a BWR, however, a PWR System is similar in concept.

- All wastes are collected and processed on a batch basis automatically after operator initiation. Normal control of the system is performed remotely from a centrally located radwaste control area.

- Most types of waste shipped offsite are in solid form. Liquid wastes and solid wastes in slurry form are solidified using a liquid-type solidification agent. Low specific activity filter/demineralizer sludges, if present, are shipped offsite in dewatered form.

- Attachment 36 shows an example of a solid waste packaging operation utilizing the liquid catalyst concept. The filling and mixing operation is performed remotely behind shielded walls. The waste to be solidified, the solidification agent and the catalyst are measured in the metering station and mixed in the drum or cask where solidification takes place.

3.5 Containment Subcompartment Pressurization

George Kowal, Manager of the Nuclear and Safety Analysis Department and Don Croneberger, Chief Structural Engineer, informed the Subcommittee of the following:

- The FLASH digital computer program has been utilized up to this time to determine the pressure response within compartments. The code has been modified to allow the input of the blowdown mass and energy release which is usually calculated independently by the reactor vendor.

- The compartment volumes or subvolumes in FLASH are modeled as nodes with boundaries determined by expected flow characteristics and special pressure differences (see Attachments 37 and 38). Where a significant pressure gradient is expected within a compartment, a subvolume approach is used.

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- Parametric studies are performed to verify the model and its physical assumptions. The flow between volumes is calculated by use of the orifice (for a small L/D) or one-dimensional momentum (for a large L/D) equation. When choked flow conditions exist, the Moody model (with a 0.6 multiplier) for maximum flow is utilized. An addition of 40% margin (20% for blowdown uncertainties and 20% for critical flow model uncertainties) is also included to produce a conservative basis for subcompartment pressurization calculations.

- Gilbert undertook the development of a rigorous multivolume program (MNODE) for subcompartment pressure calculations. This program is now in operation and is being applied to solve the AEC benchmark subcompartment problem.

- Attachment 39 illustrates a comparison between CONTEMPT and MNODE for a two volume problem.

- Attachments 40 and 41 illustrate comparisons of the semi-scale blowdown tests with RELAP-3 and MNODE.

- The analysis results as shown on Attachment 42 must be incorporated into the structural design of various subcompartments inside containment such as the reactor vessel cavity. Typically, the entire reinforced concrete structure is divided into a number of interconnected structural models for analysis purposes. The analyses performed are, for the most part, linear elastic static analyses performed with static or equivalent static loads.

- To incorporate the pressurization loadings into the analyses in combination with other loads, such as dead and live load, the peak pressures are taken from the pressurization transients and multiplied by appropriate dynamic load factors to account for dynamic effects. This transformation is illustrated in Attachment 43.

- Attachment 44 illustrates how the static design load might be combined with others in a so-called abnormal loading condition.

- A 1.4 factor has been included to account for uncertainty in predicting the pressure load.

- It is GAI's opinion at this time that the AEC-DOL position as presently stated is unduly conservative;

3.6 <u>Design for Effects of Postulated Pipe Breaks</u>

Fred Moreadith, Supervisor of Technical Services, briefed the Subcommittee and noted that:

- In a given pipe break accident situation the simultaneous effects of pressurization, jet impingement, pipe whip, flooding and corresponding degradation of environment must be considered in the design process.

- To solve most of the potential problems during the layout stage of a project they apply the concepts of isolation, separation, redundancy, and enclosure. Where such approaches cannot be employed, individual pipe restraints and jet impingement shields are utilized.

- GAI has conducted basic structural analytical work to investigate some of the parameters believed to be important during the pressure and pipe whip transients associated with postulated pipe break accidents. The results and discussion are related to modeling techniques, to the effect of varying the thrust rise time, to the gap between restraint and pipe, and to load combinations.

- The design conditions associated with postulated pipe break accidents have received considerable attention within Gilbert Associates during the past several years and they expect to continue to expend considerable effort on these kinds of problems in the future.

3.7 Energy Parks

Ken Broome, Chief Civil Engineer of the Utilities Division, informed the Subcommittee of the following:

- There are three basic questions concerning parks -
  - What are the capacity limitations?
  - How closely spaced should multiple units be in such a development?
  - How close can related industrial and residential facilities be located?

- Park capacity limitations appear to be limited by the cooling water supply.

- The cumulative radiation dose from normal plant operations and turbine missiles would affect spacing.

- Current guidelines on population concentration require a two-mile no population zone and 30,000 limit within five miles. A park could conceivably have a construction force of 3,000 to 5,000 which could develop a population of 25,000.

In response to a question from Mr. Bender concerning the possible size of parks, Broome indicated that they have heard of needs in the order of 20,000 megawatts.

3.8 Seismic Considerations

Don Croneberger, Chief Structural Engineer, informed the Subcommittee of the following:

- GAI engineers are actively participating either as members or observers in various industrial committees, e.g., IEEE Working Group 2.5 on the Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations, Seismic Task Group of ASCE Nuclear Structure and Material Committee, and ASME Task Group on Dynamic Analysis.

- The latest contribution by Gilbert has been to define the statistical independence of three component artificial time histories. Gilbert's approach was to calculate the correlation coefficients of the component strong motion accelerograms recorded in the western part of the U.S. They calculated the mean values and standard deviations of these correlation coefficients. Based on these studies, they have concluded that the artificially-generated time histories should have absolute correlation coefficients equal to or less than 0.16.

In response to a question from Mr. Bender concerning the upper limit for design ground acceleration, Croneberger indicated that without major modifications to their concepts, the upper limit is around two-thirds to three-quarters the acceleration of gravity.

3.9 Combustible Gas Control

George Kowal, Manager of the Nuclear and Safety Analysis Department, informed the Subcommittee of the following:

- General Design Criterion 41 requires that systems be provided to control the concentrations of hydrogen, oxygen, and other substances which may be released into the reactor containment following postulated accidents to assure that containment integrity is maintained. AEC Regulatory Guide 1.7 describes an acceptable method of implementing this criterion for light water reactors.

- The current draft of Regulatory Guide 1.7 requires a 0.2 mil metal reaction in 120 seconds as the criterion for the volume of combustible gas generated. This volume is an order of magnitude lower than the requirements of the original guide and the time duration imposes restrictive design requirements in new generation BWR's.

- During the drywell depressurization phase which occurs approximately 4 to 5 minutes after a LOCA, the hydrogen concentration increases as the drywell pressure decreases. Since all the hydrogen generated by the metal-water reaction is already in the drywell, without an elaborate system, the concentration may exceed the guideline values for safe limits. If the time for the reaction were extended, even though the same total hydrogen were produced, the control could be simplified.

- o The ECCS by virtue of its design and redundancy limits the clad temperature to a design value far below the 2200°F of the interim criterion. GAI feels the calculated design value of temperature should be used. This allows them a considerable time margin and, they believe, a more realistic approach.

The meeting was adjourned at 5:00 p.m.

******************

| GAI EMPLOYEE | TITLE |
|---|---|
| Broome, K. | Chief Civil Engineer |
| Chen, C. | Research Engineer |
| Croneberger, D. K. | Chief Structural Engineer |
| Goodman, S. D. | Engineering Manager - Utilities Division |
| Hottenstein, E. R. | Project Manager |
| Kowal, G. M. | Department Manager, Nuclear & Safety Analysis |
| Lorenz, H. F. | Vice President & General Manager, Utilities Division |
| Mathys, R. | Manager Design Control |
| McMahon, T. M. | Chief Electrical Engineer |
| Meek, W. E. | Chief Mechanical/Nuclear Engineer |
| Moreadith, F. | Superviser of Tech. Services, Structural Dept. |
| Myers, L. B. | Nuclear Engineering - Licensing |
| Pflum, W. F. | Supervising Electrical Engineer |
| Porter, H. R. | Engineering Manager - Utilities Division |
| Raquet, D. A. | Security Engineer |
| Sailer, W. F. | Project Manager |
| Shields, W. B. | Engineering Manager - Utilities Division |
| White, D. P. | Supervising Engineer Mechanical/Nuclear Dept. |
| Willems, V. H. | Project Instrument Engineer |

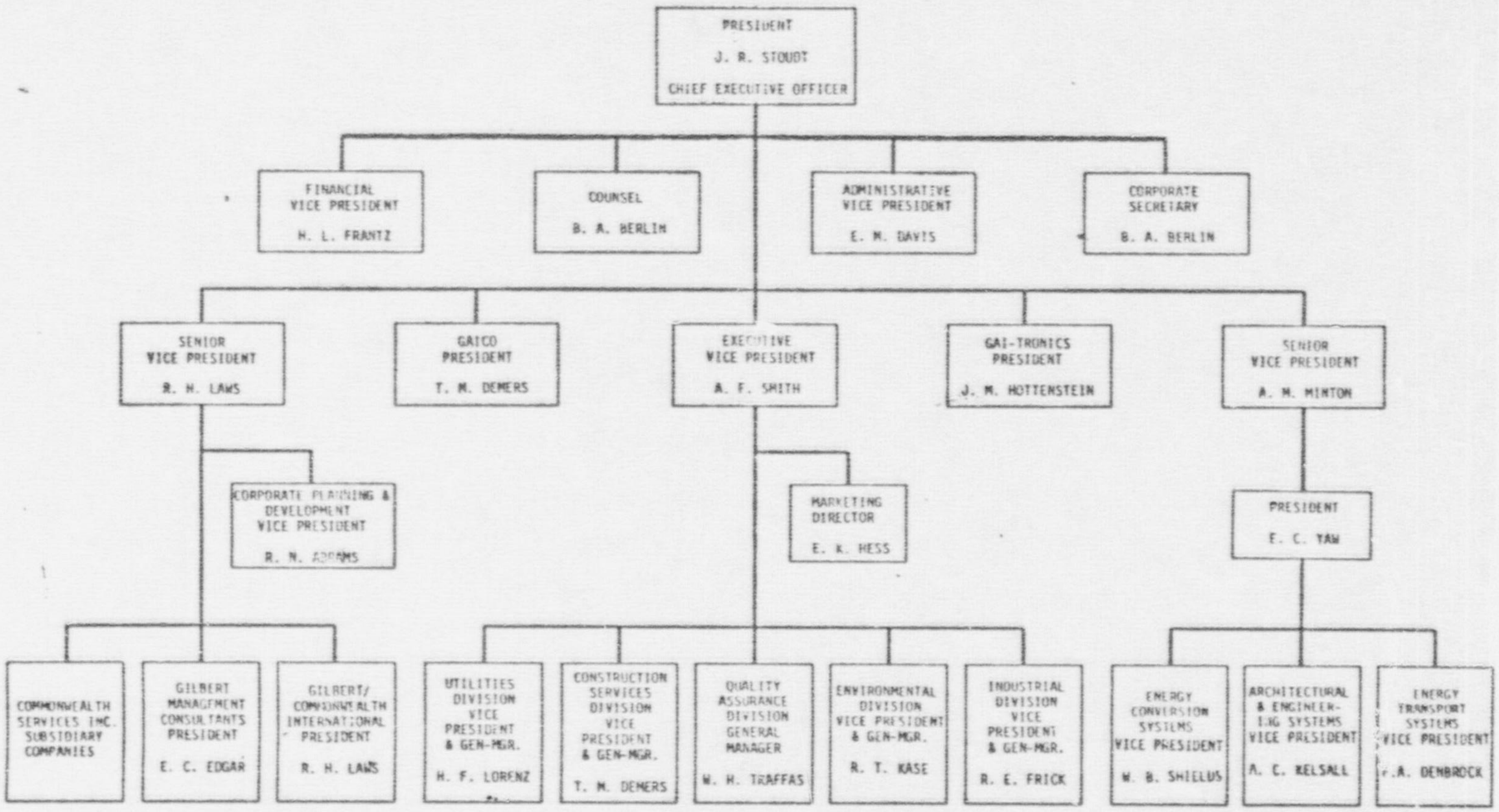
Rev. 3 - 8/14/74 Allen

FINAL
AGENDA

ACRS SUBCOMMITTEE

August 23, 1974

| | ITEM | | PARTICIPANT | ALTERNATES | PREPARED STATEMENT TIME ALLOTMENT |
|---|---|---|---|---|---|
| 1. | Role of the Architect-Engineer | Corporate/ Utility Division | H.F. Lorenz | H.R. Porter | 10 min. |
| | | Commonwealth | W.B. Shields | | |
| 2. | Standardization | General Design a | S.D. Goodman | | 5 min. |
| | | Control/Assurance b | R. Mathys | | 5 min. |
| | | Pre-Designated site Criteria c | S.D. Goodman | | 5 min. |
| | | Std. Plant d | E.R. Hottenstein | | 5 min. |
| 3. | Interfaces with NSS Supplier | Custom and Std. Plant | E.R. Hottenstein | | 5 min. |
| 4. | Effects of Operating Experience | | W.F. Sailer | W.E. Meek | 5 min. |
| 5. | R&D in A-E Areas | | ~~K. Broome~~ W. Sh... | G. Kowal | 5 min. |
| 6. | Regulatory Guides | General | L. Myers | | 5 min. |
| 7. | Turbine Generated Missiles | Analysis | G. Kowal | | 4 min. |
| | | Design | D. Croneberger | F. Moreadith | 4 min. |
| 8. | Qualification of Equipment to Operate in the Post-LOCA Environment | Elect./I&C | T. McMahon | W. Pflum<br>V. Willems | 5 min. |
| 9. | Radwaste Management | | D. White | | 4 min. |
| 10. | Designs to Improve Industrial Security (closed session if necessary) | | A. Raquet | | 4 min. |
| 11. | Containment Subcompartment Pressurization | Analysis a | G. Kowal | | 4 min. |
| | | Design b | D. Croneberger | | 4 min. |
| 12. | Other Generic Items | | | | |
| | Pipe Break/whip problems | Structural Dept. a | D. Croneberger | F. Moreadith | 4 min. |
| | Energy parks | Civil Dept. b | K. Broome | | 4 min. |
| | Structural Seismic Analysis | Structural Dept. c | D. Croneberger | C. Chen | 4 min. |
| | Personnel Exposure Data Avail. | Nuclear Anal. d | G. Kowal | | 4 min. |
| | Hydrogen Recombiner Reg. Guide Issue | Nuc. Anal. e | G. Kowal | | 4 min. |
| | Heat Sink Cooling Water | Civil Dept. f | K. Broome | | 4 min. |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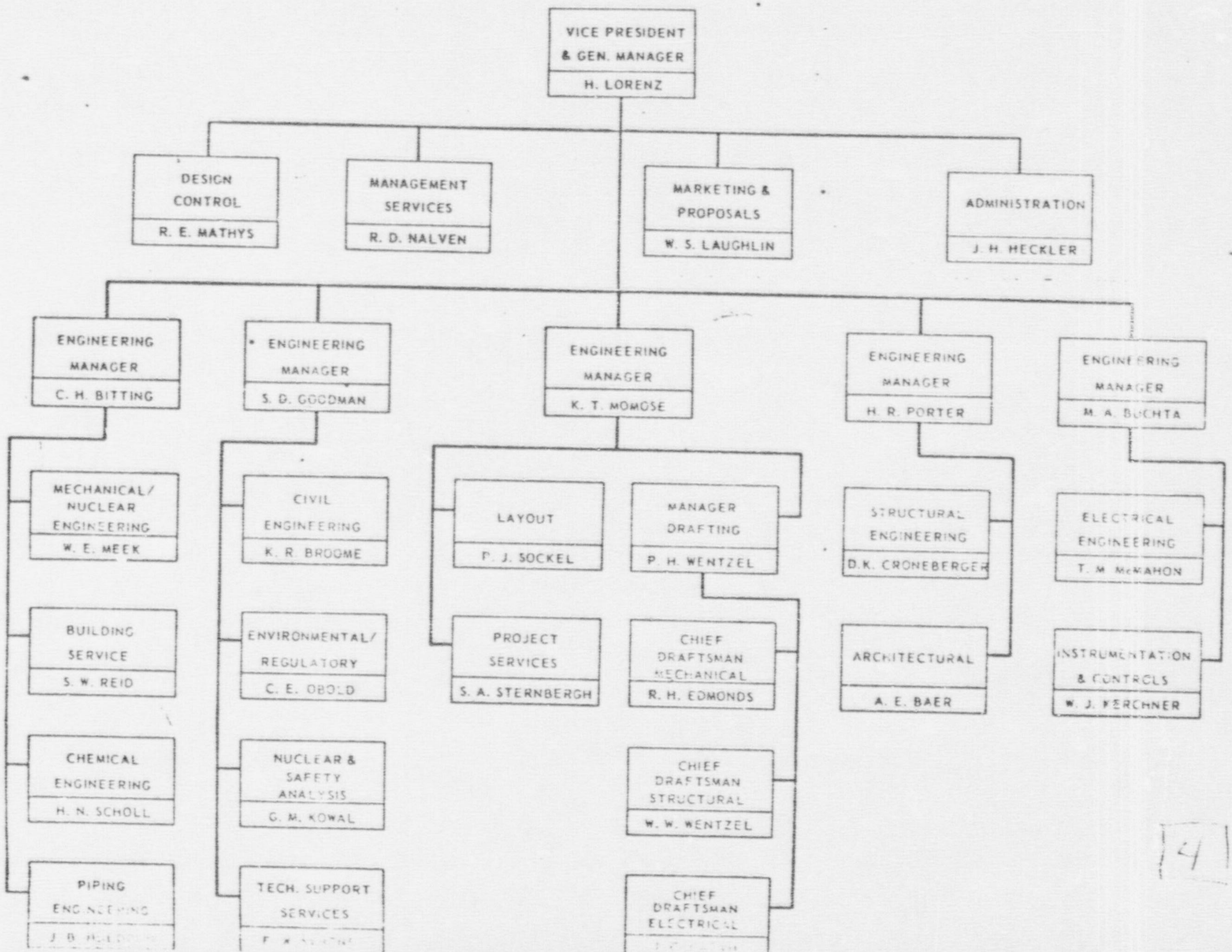


J

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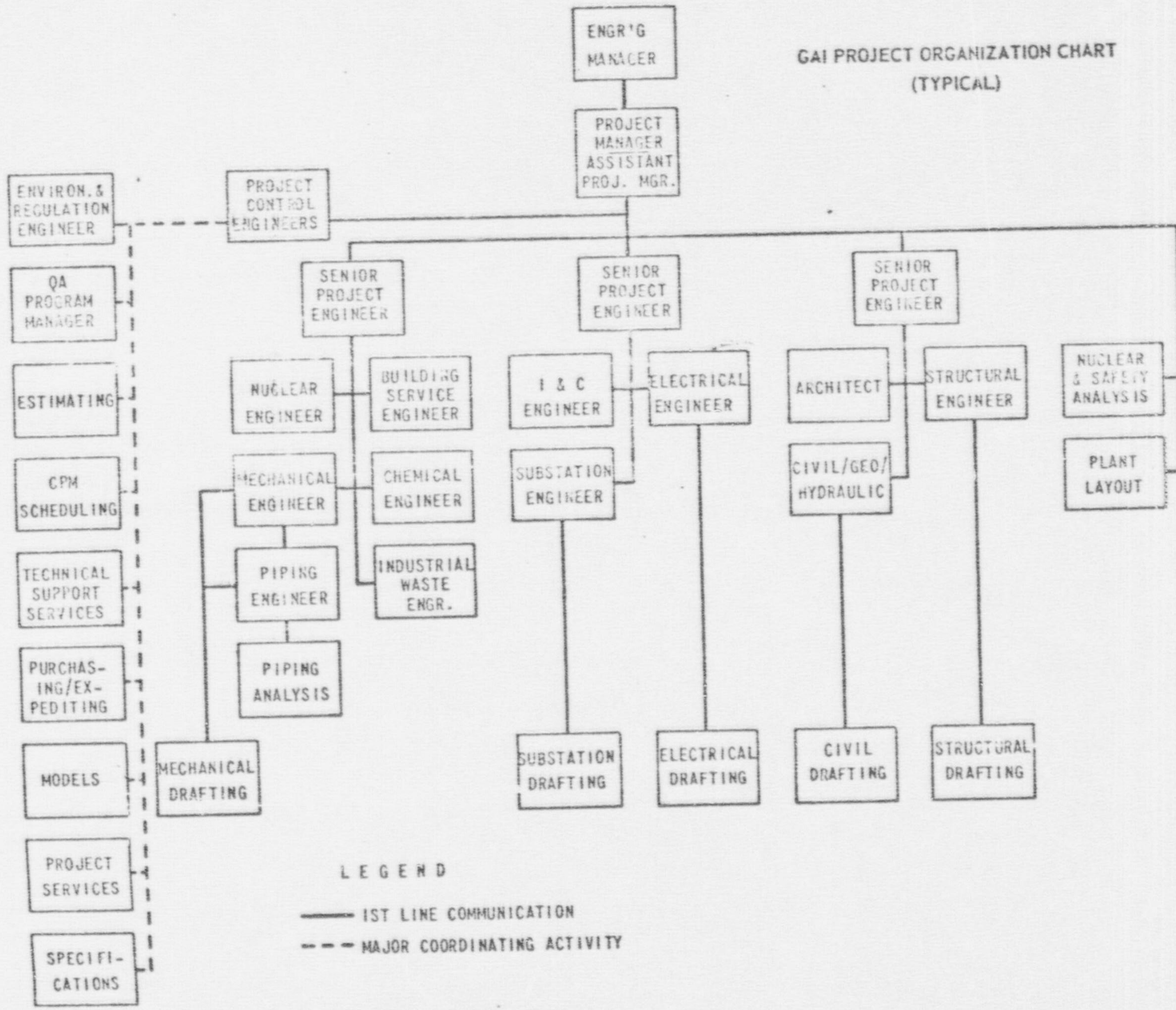
5

## GAI PROJECT ORGANIZATION CHART
## (TYPICAL)



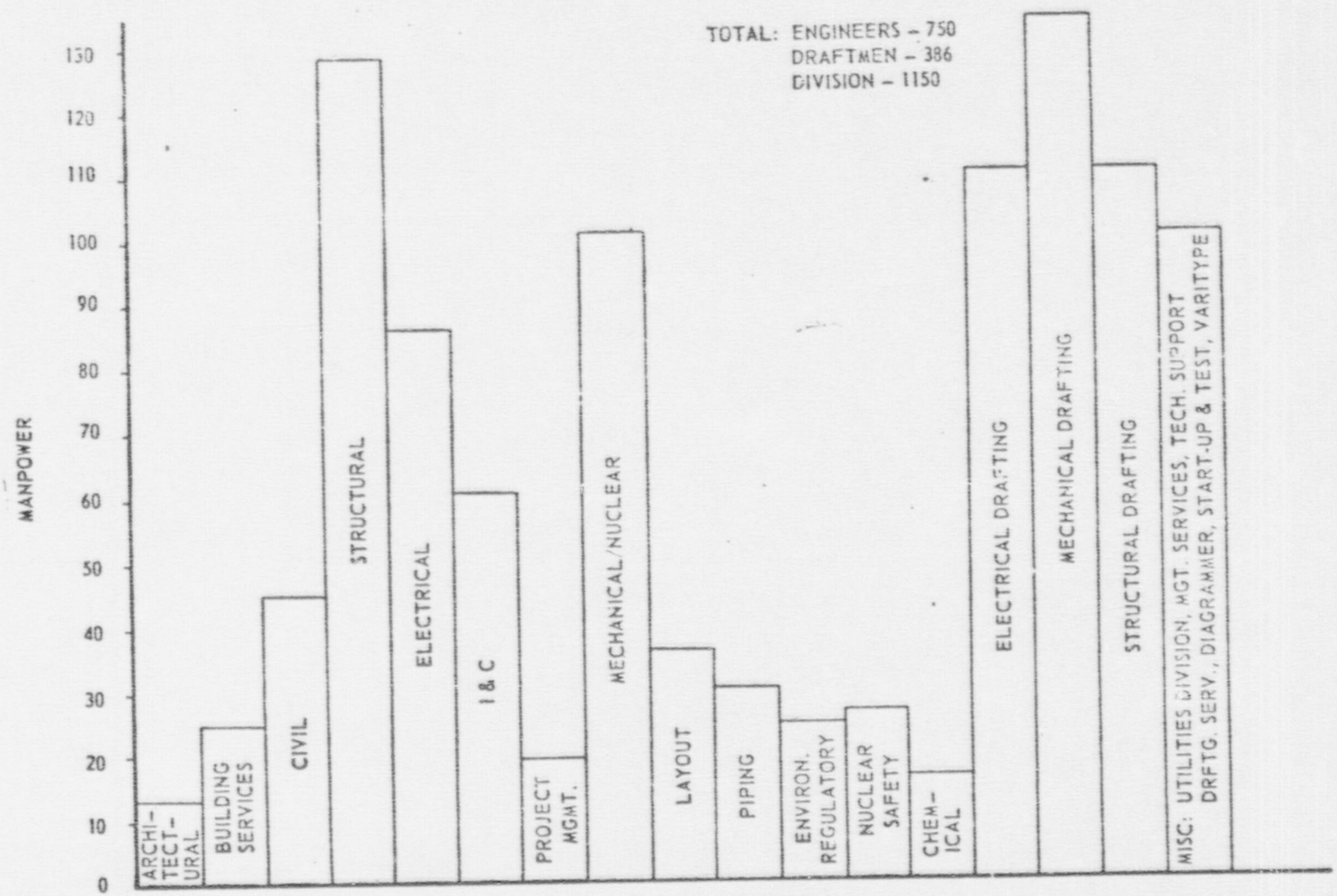

LEGEND

——— 1ST LINE COMMUNICATION

- - - MAJOR COORDINATING ACTIVITY

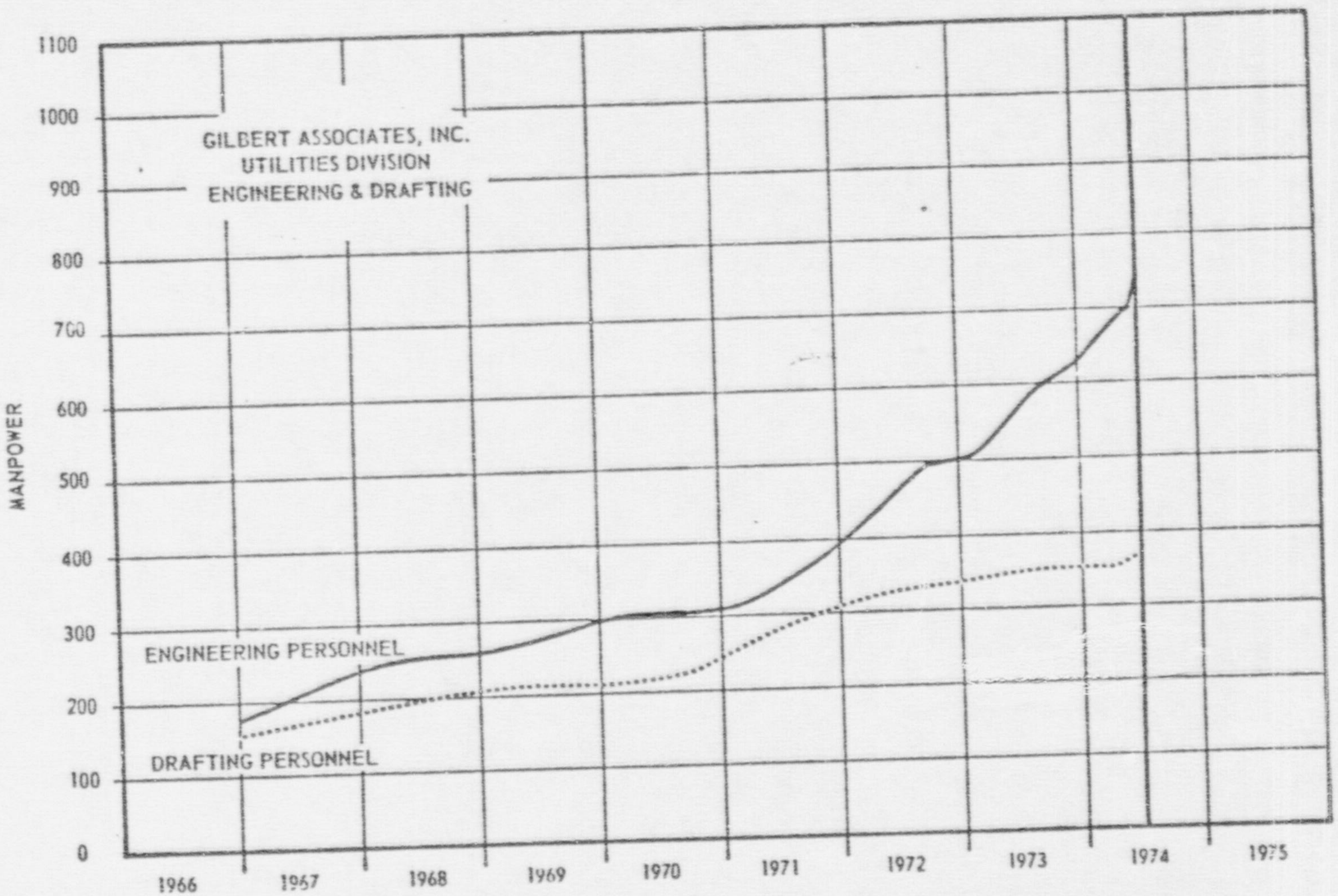
5

5




DEPARTMENT PERSONNEL
UTILITIES DIVISION

7


GILBERT ASSOCIATES, INC.
UTILITIES DIVISION
ENGINEERING & DRAFTING
MANPOWER
1100
1000
900
800
700
600
500
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300
200
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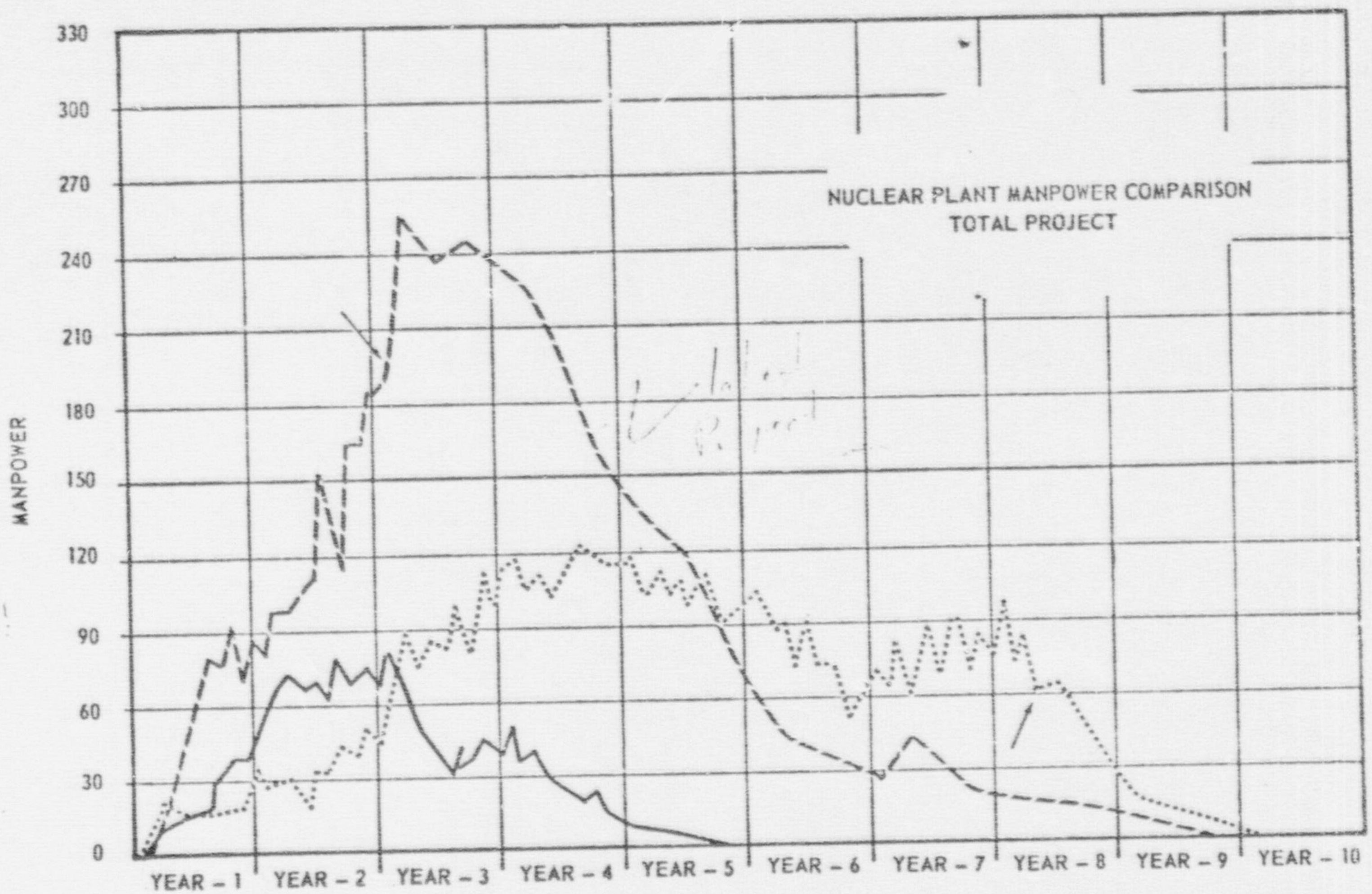
ENGINEERING PERSONNEL
DRAFTING PERSONNEL
1966
1967
1968
1969
1970
1971
1972
1973
1974
1975

7


NUCLEAR PLANT MANPOWER COMPARISON
TOTAL PROJECT
MANPOWER
330
300
270
240
210
180
150
120
90
60
30
0
YEAR – 1
YEAR – 2
YEAR – 3
YEAR – 4
YEAR – 5
YEAR – 6
YEAR – 7
YEAR – 8
YEAR – 9
YEAR – 10

8

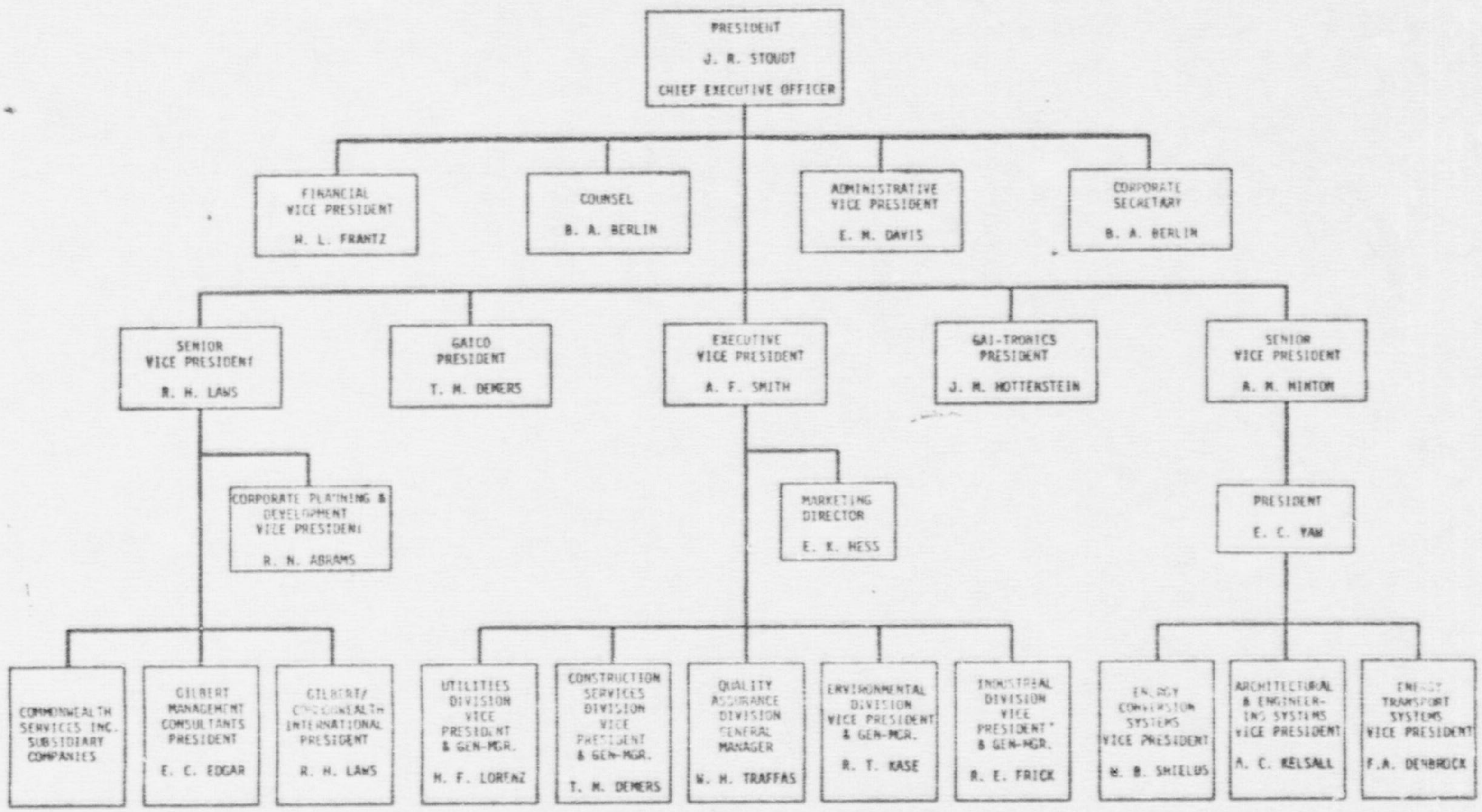
6

GAI
UTILITIES DIVISION
MAJOR NUCLEAR DESIGN EFFORTS

| PROJECT | SIZE MW | STATUS/C.O. |
|---|---|---|
| SAXTON | | COMPLETED |
| GINNA | 490 | COMPLETED |
| MIHAMA #1 | 485 | COMPLETED |
| TAKAHAMA #1 | 815 | COMPLETED |
| THREE MILE ISLAND #1 | 840 | COMPLETED |
| CRYSTAL RIVER #3 | 855 | 3/1/75 |
| KO-RI #1 | 600 | 12/1/75 |
| KRSKO | 600 | 2/1/79 |
| KO-RI #2 | 600 | 10/1/79 |
| OHI UNITS 1 & 2 | 1100 (2) | 11/1/76, 5/1/77 |
| V.C. SUMMER #1 | 915 | 5/1/78 |
| PERRY UNITS 1 & 2 | 1200 (2) | 6/1/79, 6/1/80 |
| ERIE UNITS 1 & 2 | 1100 (2) | 1982, 1984 |
| MULHEIM | 1300 | 1978 |

7

10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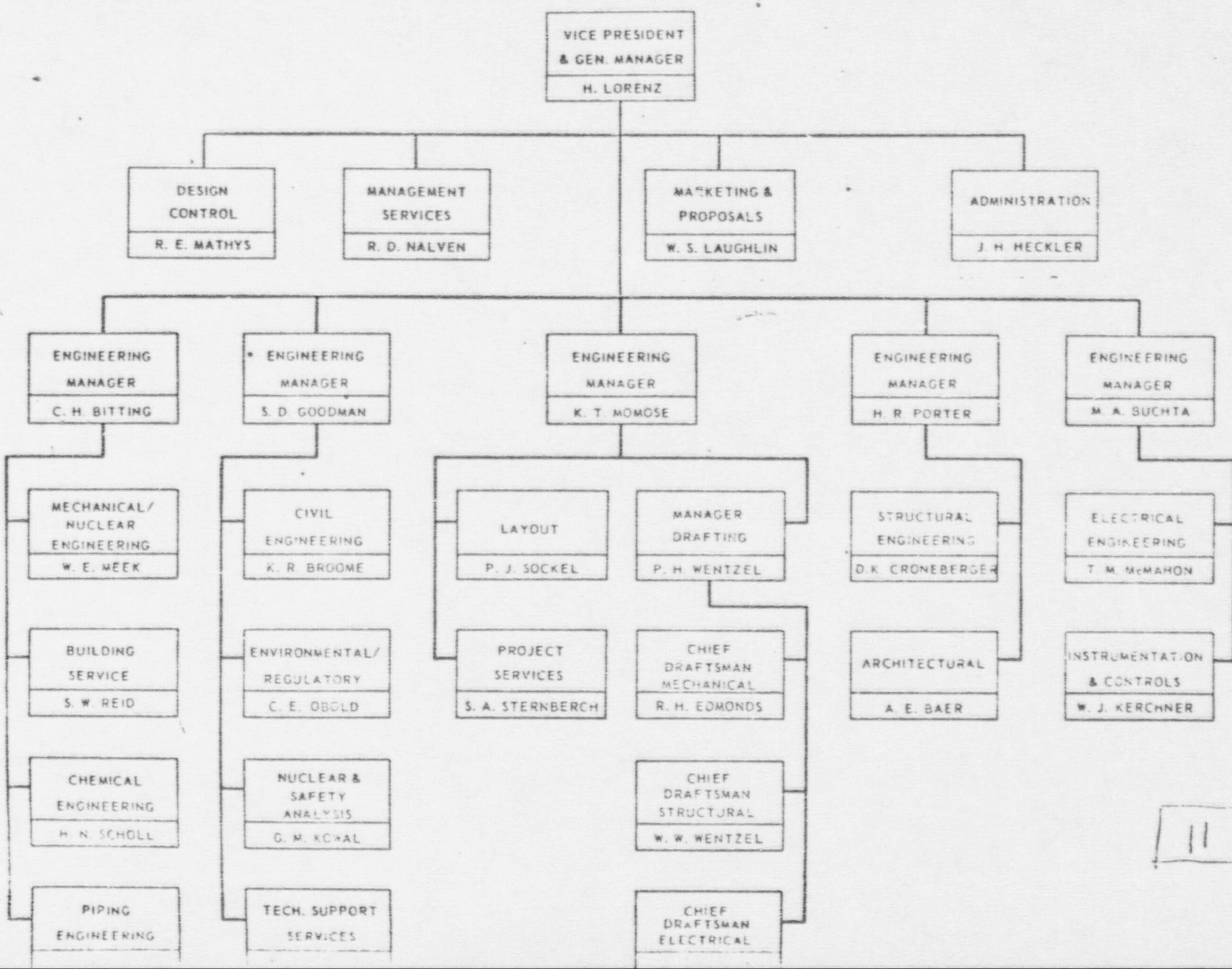


21

## STANDARD PLANT

## OVERALL SCHEDULE

| YEAR | 1974 | 1975 | 1976 | LAPSE TIME (MONTHS) |
|---|---|---|---|---|
| I. PHASE I - Conceptual Design Phase | | | | |
| START | June | | | 4 |
| FINISH | October | | | |
| II. PHASE II - BOPSSAR Development | | | | |
| START | September | | | 10 |
| FINISH | | June | | |
| III. PHASE III - Postdocketing Review | | | | |
| START | | June | | 14* |
| FINISH | | | August | |
| | | Total | | 28 |

* SER Supplement Filed

12

STANDARD PLANT

PHASE I

BASIC OBJECTIVES

I. DEVELOPMENT OF CRITERIA

II. DEVELOPMENT OF PLANT CONCEPTS

III. NSSS - SCOPE INTERFACE(S) RESOLUTION

IV. ACCEPTANCE REVIEWS

V. REPORT - PHASE I ACTIVITIES

## STANDARD PLANT

## PHASE I

## BASIC OBJECTIVES

I. DEVELOPMENT OF CRITERIA:

A. GENERAL

1. SITE - RELATED

2. PLANT - RELATED

B. SPECIFIC

1. NSSS

2. L-BOP*

II. DEVELOPMENT OF PLANT CONCEPTS:

A. COMMONALITY FEASIBILITY STUDIES

1. CONTAINMENT

2. L-BOP*

* L-BOP - Limited in that it only includes reactor, auxiliary, control, and fuel handling buildings.

III. NSSS - SCOPE INTERFACE(S) RESOLUTION:

A. L-BOP - TICK, TACK, TOE

B. CONCEPTUAL - FLOW DIAGRAMS

C. OTHER AREAS

IV. ACCEPTANCE REVIEWS:

A. AEC

B. CAI-GAI ENGINEERING DEPTS.

C. NSSS SUPPLIERS

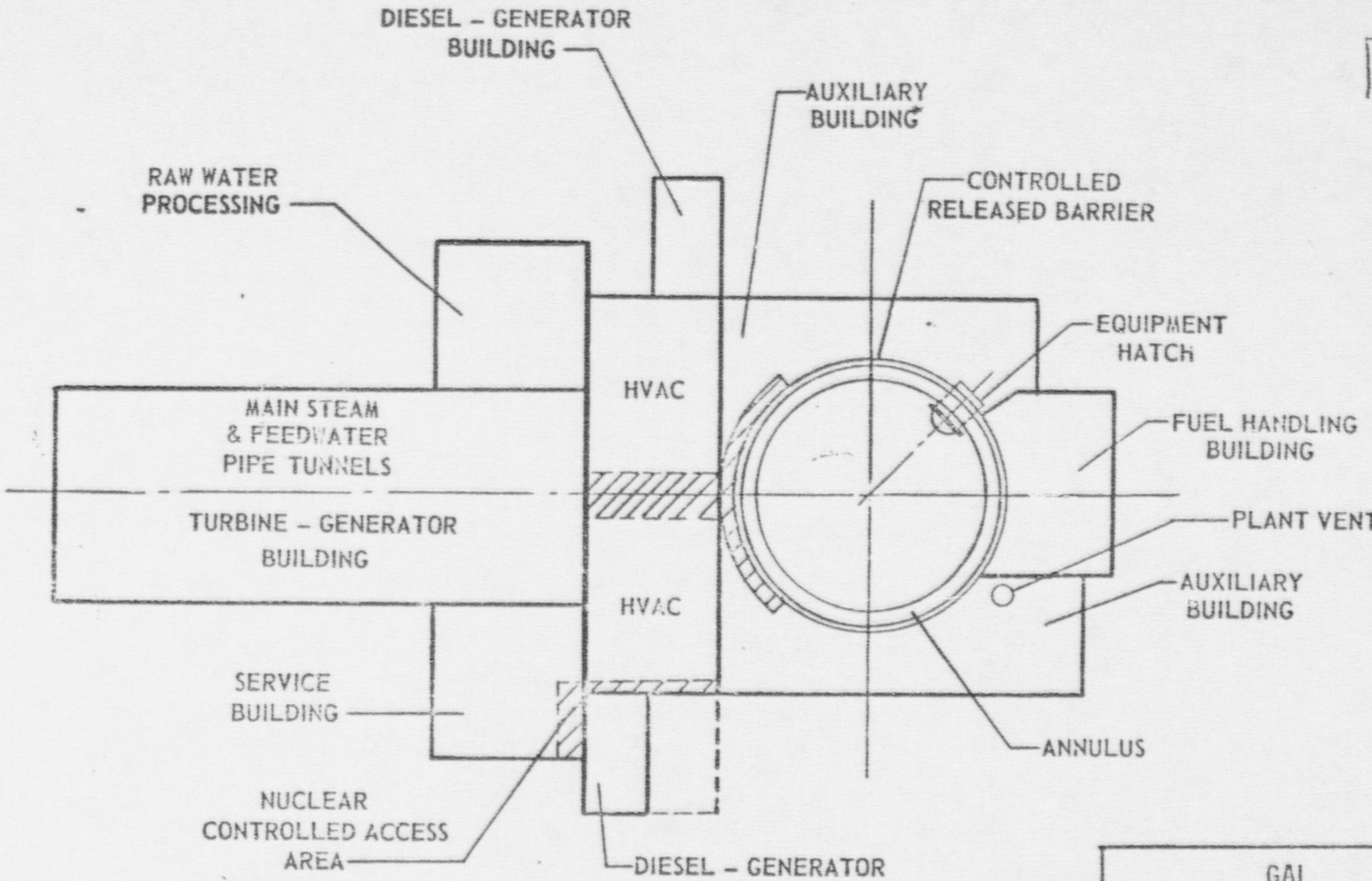
D. UTILITY CLIENTS

E. GAI MANAGEMENT

V. REPORT - PHASE I ACTIVITIES:

A. GAI - STANDARD PLANT GROUP

91

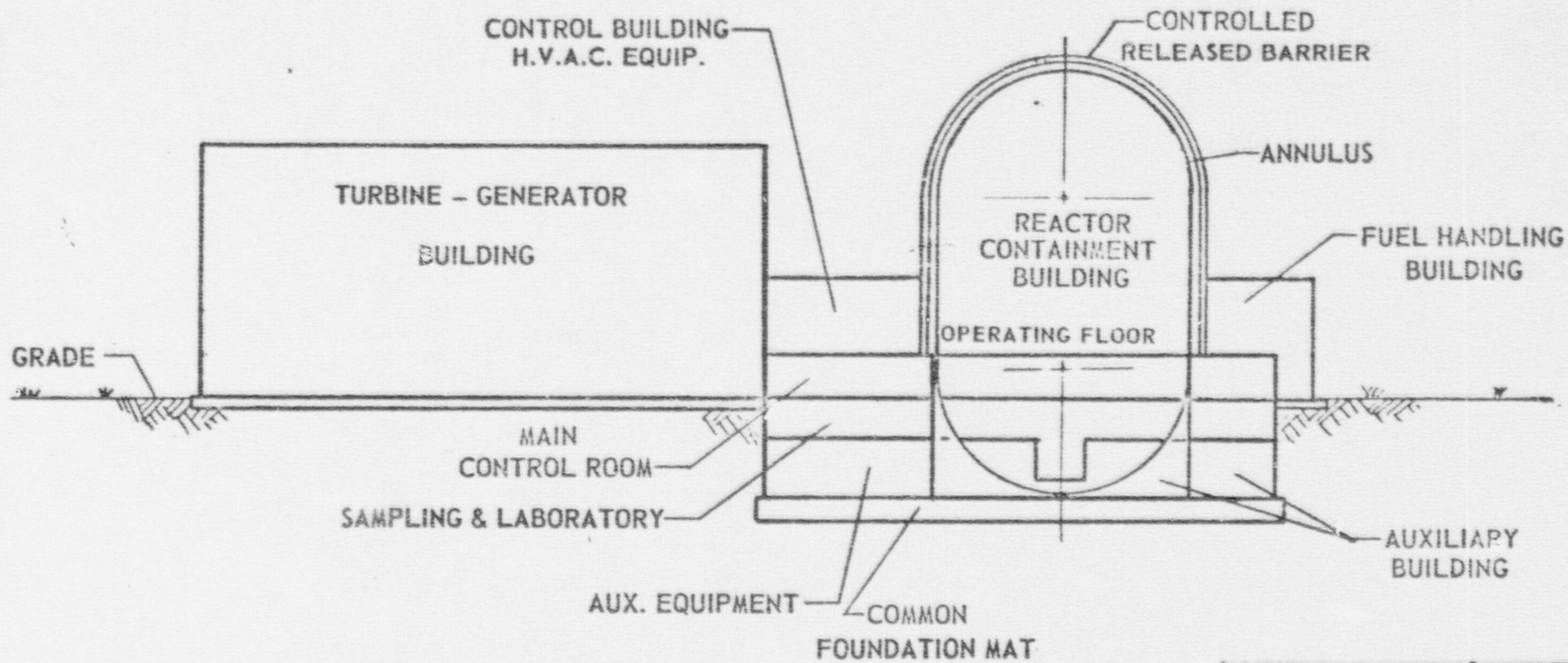



GAI
STANDARD PLANT
CONCEPTS

PLAN
CYLINDRICAL CONTAINMENT

EXHIBIT NO. 1

17



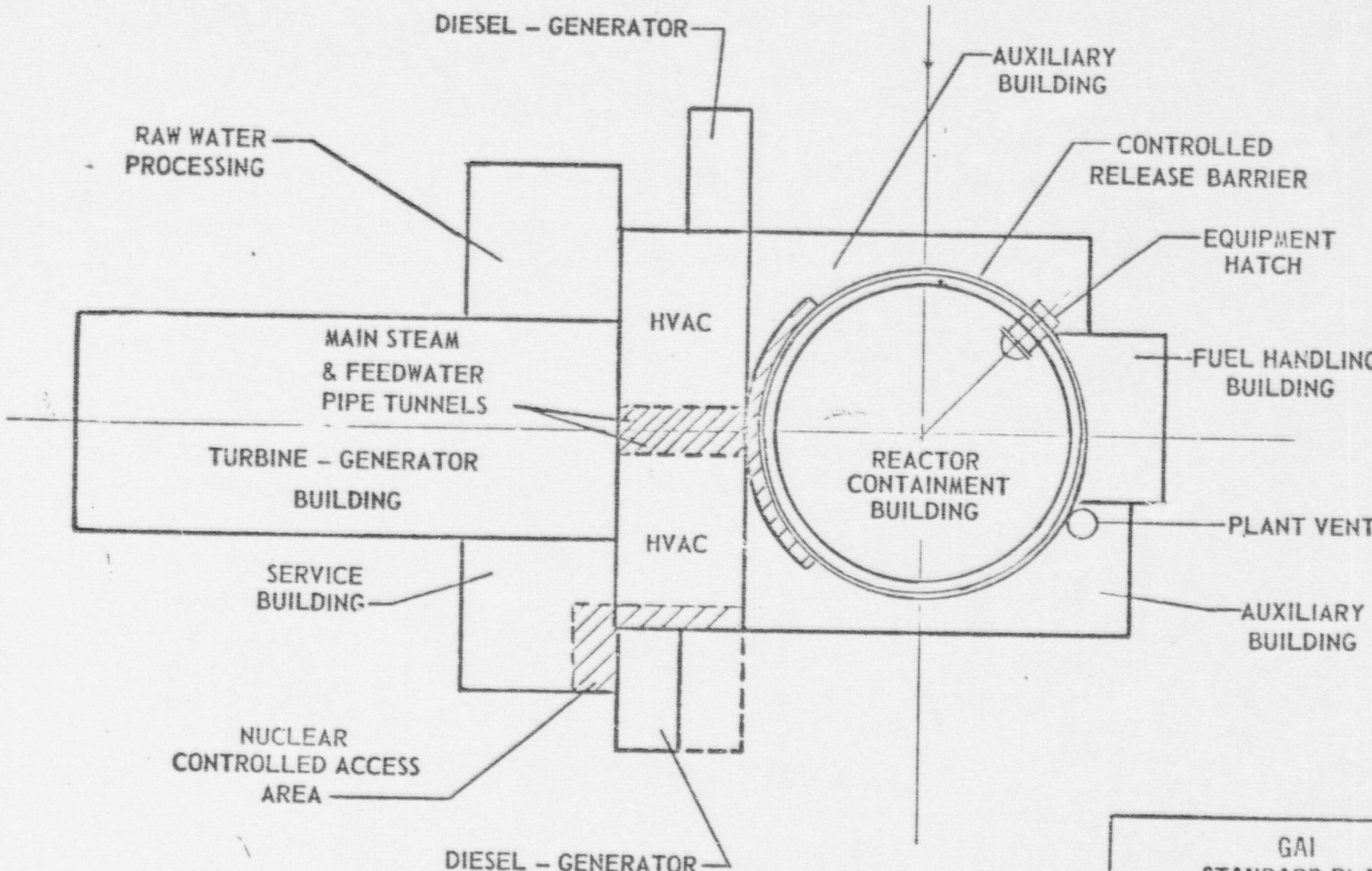

GAI
STANDARD PLANT
CONCEPTS

SECTION THRU
CYLINDRICAL CONTAINMENT

EXHIBIT NO. 2

17

13

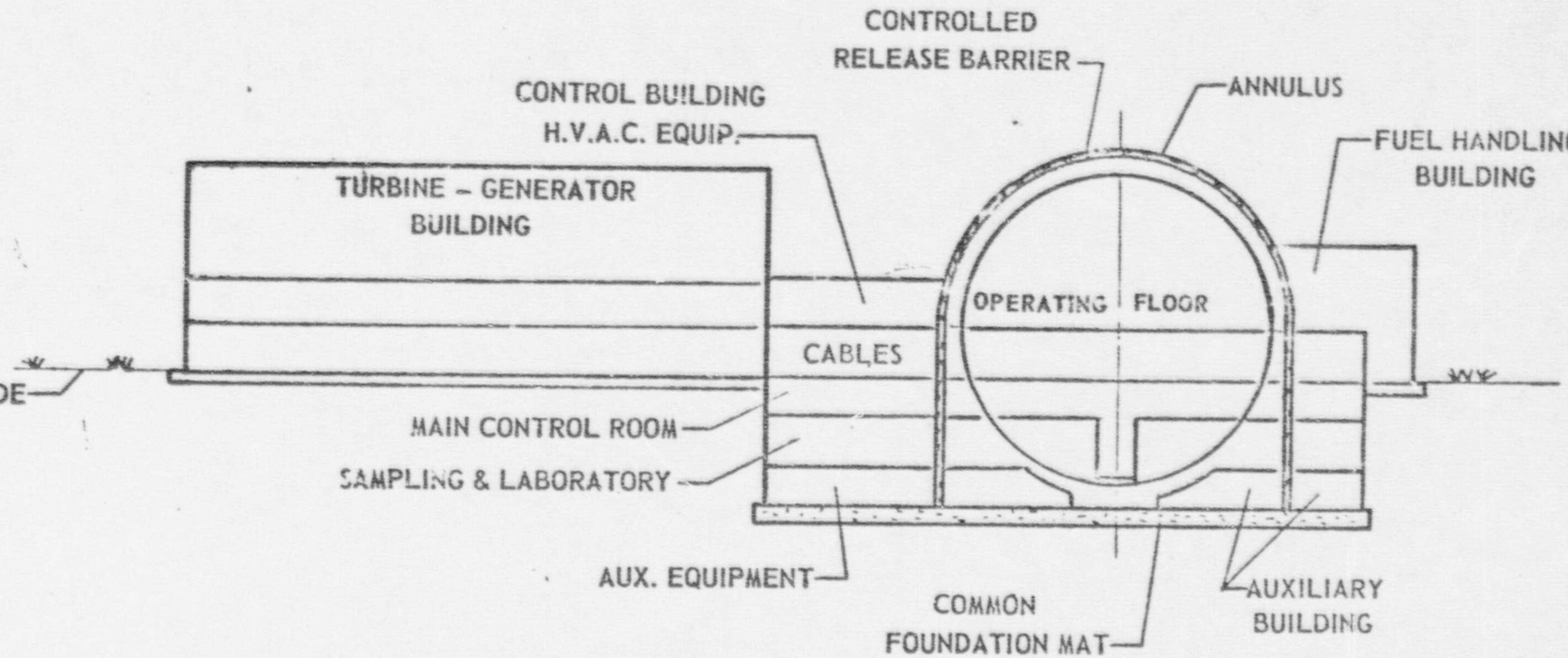



GAI
STANDARD PLANT CONCEPTS

PLAN
SPHERICAL CONTAINMENT

EXHIBIT NO. 3

19




GAI
STANDARD PLANT
CONCEPTS

SECTION THRU
SPHERICAL CONTAINMENT

EXHIBIT NO. 4

## STANDARD PLANT

## NSSS - SYSTEM SCOPE SPLIT

I. PWR-NSSS SUPPLIERS SCOPE - BY SYSTEM:

A. REACTOR COOLANT

B. EMERGENCY CORE COOLING

C. EMERGENCY BORATION

D. RESIDUAL HEAT REMOVAL

E. INCORE FAILED FUEL DETECTION

F. ENGINEERED SAFETY FEATURES INSTRUMENTATION & CONTROL

G. REACTOR PROTECTION INSTRUMENTATION & CONTROL

H. NUCLEAR STEAM SUPPLY SYSTEM INSTRUMENTATION & CONTROL

I. FUEL HANDLING EQUIPMENT & FUEL STORAGE RACKS

STANDARD PLANT

L-BOP - SYSTEM SCOPE SPLIT

II. GAI - L-BOP - STANDARD PLANT SCOPE - BY SYSTEM:

A. CHEMICAL & VOLUME CONTROL

B. BORON RECOVERY

C. RADIOACTIVE WASTE MANAGEMENT

D. TURBINE BYPASS & STEAM DUMP

E. SPENT FUEL POOL COOLING & PURIFICATION

F. POST-LOCA CONTAINMENT COOLING & IODINE REMOVAL

G. STEAM GENERATOR BLOWDOWN PROCESSING

H. POST-LOCA HYDROGEN RECOMBINER

I. COMPONENT COOLING WATER

J. HEATING, VENTILATION & AIR CONDITIONING

K. CONTAINMENT ISOLATION

L. EMERGENCY FEEDWATER

M. SAMPLING SYSTEMS

N. STATION AND INSTRUMENT AIR

O. DIESEL GENERATORS

## TICK, TACK, TOE FOR INTERFACE

### RESOLUTION, L-BOP*

| EXAMPLE | SYSTEM | CRITERIA | | DESIGN | | | ANALYSIS |
|---|---|---|---|---|---|---|---|
| | | OPERATION PARAMETERS | CODES & STANDARDS | SYSTEM FUNCTIONAL LOGIC | FUNCTIONAL SPECS. | DETAILED DESIGN | |
| NO. 1 | EMERGENCY CORE COOLING | | | | | | |
| | EQUIPMENT | NSSS | NSSS | NSSS | NSSS | GAI | NSSS |
| | BALANCE OF SYSTEM | NSSS | NSSS | NSSS | GAI | GAI | GAI |
| NO. 2 | RAD. WASTE MANAGEMENT | | | | | | |
| | EQUIPMENT | GAI & NSSS | GAI | GAI | GAI | GAI | GAI |
| | BALANCE OF SYSTEM | GAI & NSSS | GAI | GAI | GAI | GAI | GAI |

* L-BOP - Limited in that it only includes reactor, auxiliary, control, and fuel handling buildings

4.1

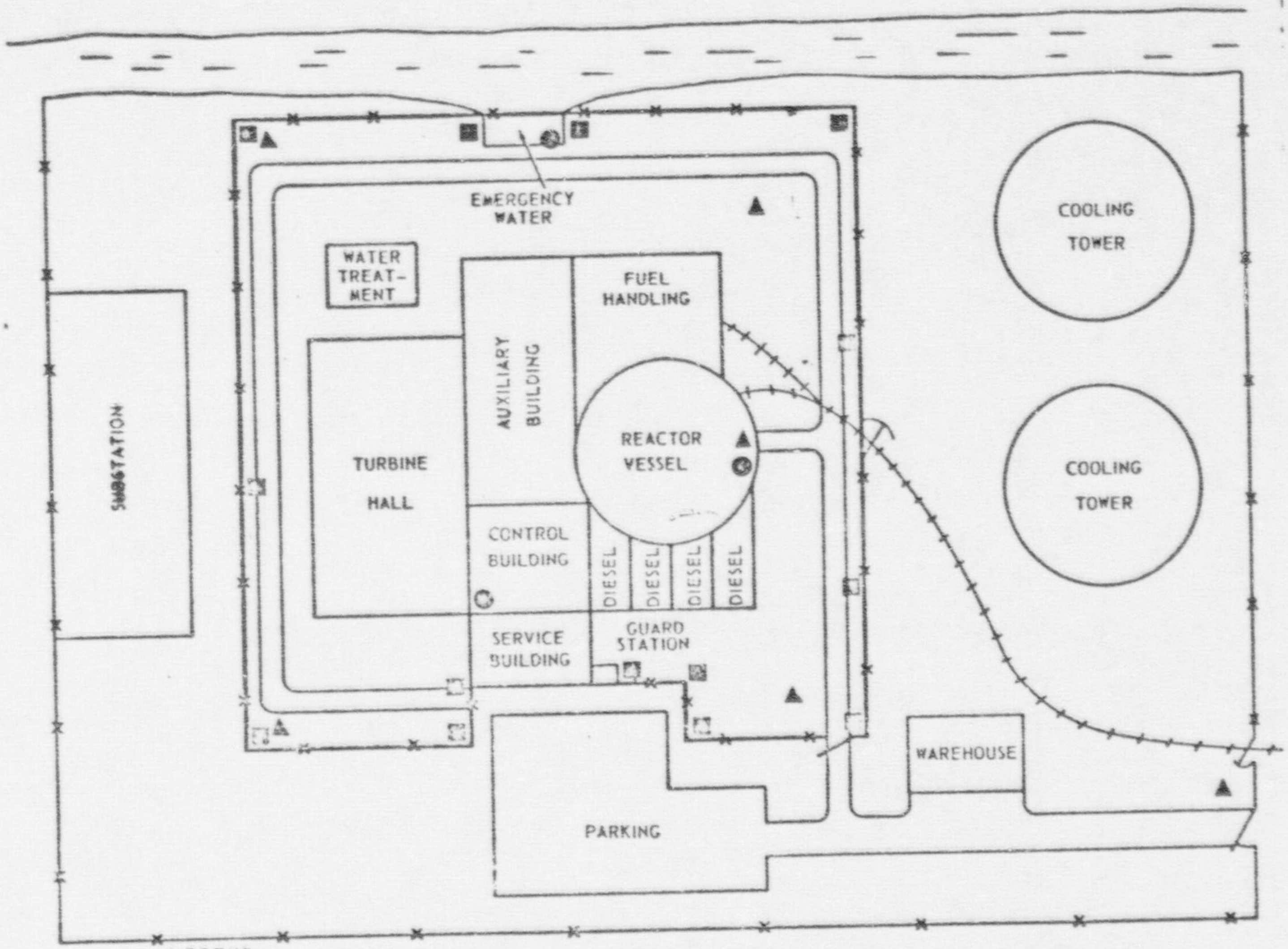
## SOURCES OF OPERATING INFORMATION

- REACTOR OPERATING EXPERIENCE REPORTS (ROE'S)
- REACTOR CONSTRUCTION EXPERIENCE REPORTS
- REGULATORY OPERATIONS BULLETINS
- CLIENT MONTHLY OPERATING REPORTS
- NUCLEAR POWER EXPERIENCE (VERNA REPORTS)
- EEI REPORTS
- CONFERENCES

## ADDITIONAL INFORMATION SOURCES

- STARTUP/TEST PERSONNEL
- CONTINUING SERVICES PROJECTS
- UTILITY OPERATING AND MAINTENANCE STAFFS
- DESIGN ENGINEERS ASSIGNED TO OPERATING PLANTS FOR TRAINING

FIGURE 1
TYPICAL PLANT SECURITY LAYOUT




LEGEND

- TV
- INTRUSION DETECTION ZONES
- INDOOR INTRUSION DETECTION

25

FIGURE 2
PLACEMENT OF SAFETY EQUIPMENT
FOR SPHERICAL CONTAINMENT

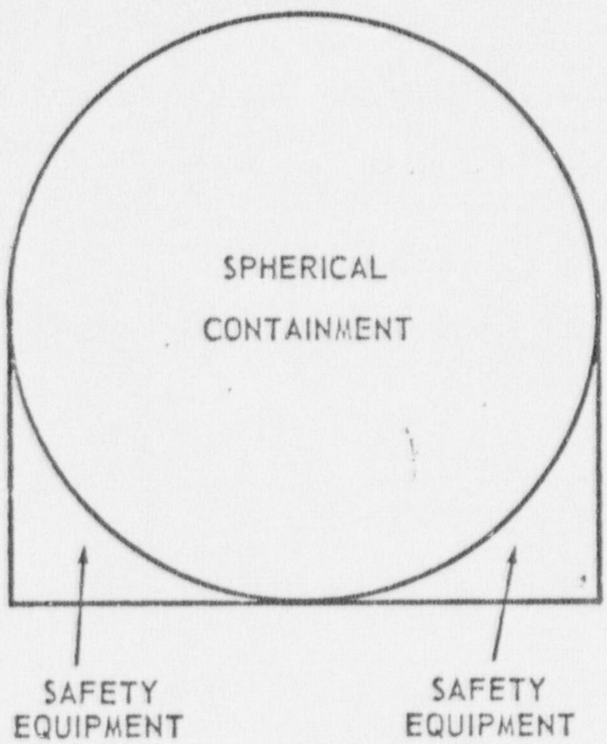
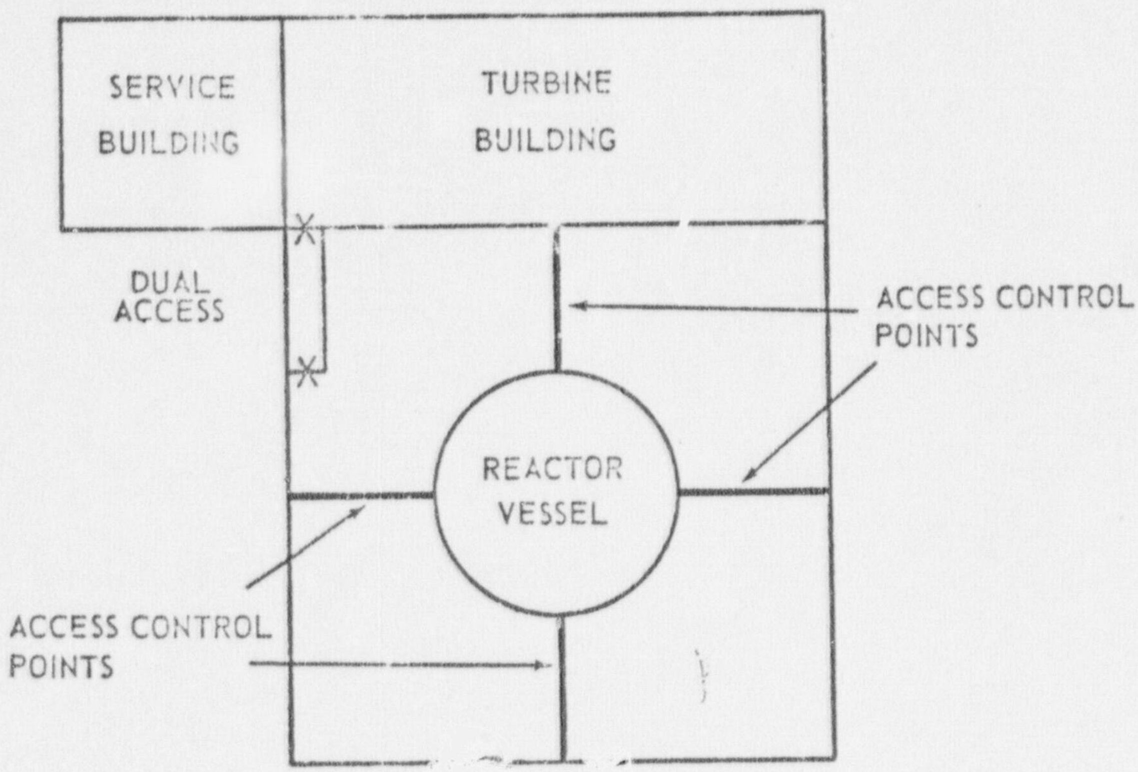


FIGURE 3
TYPICAL PLANT ACCESS CONTROL



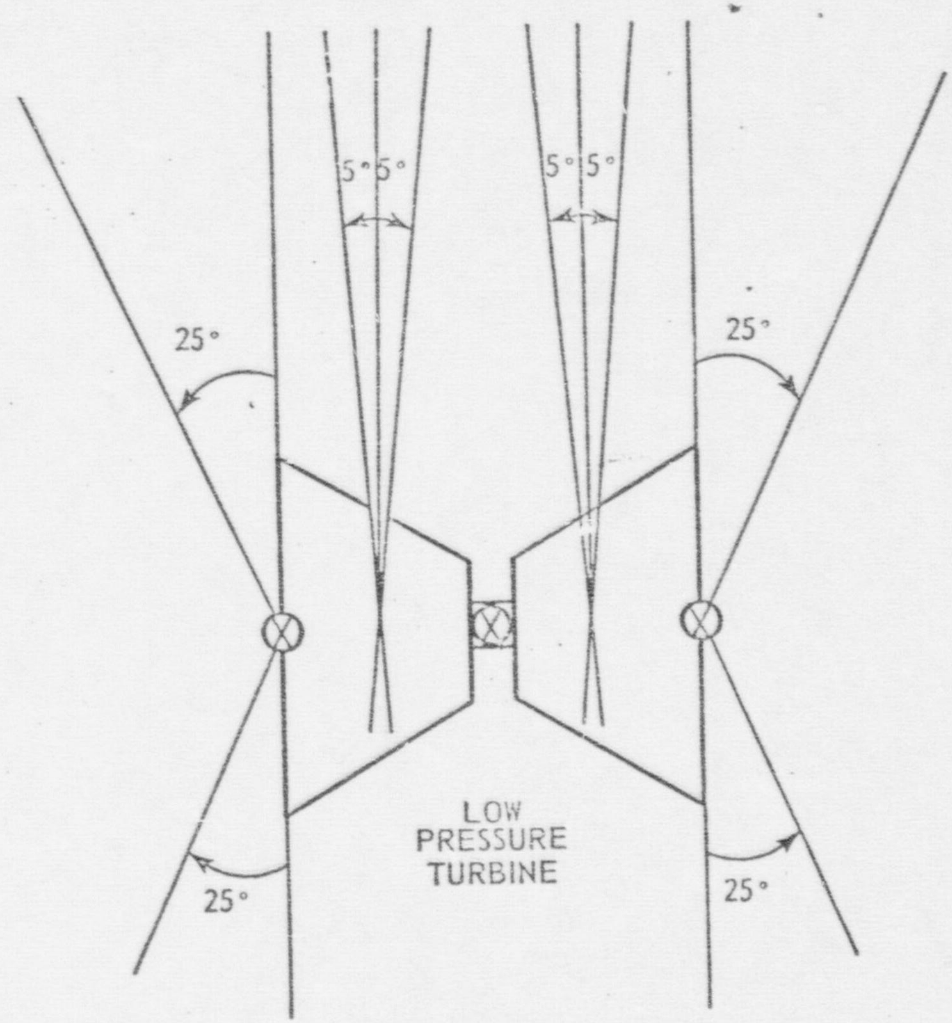

26 A

26B

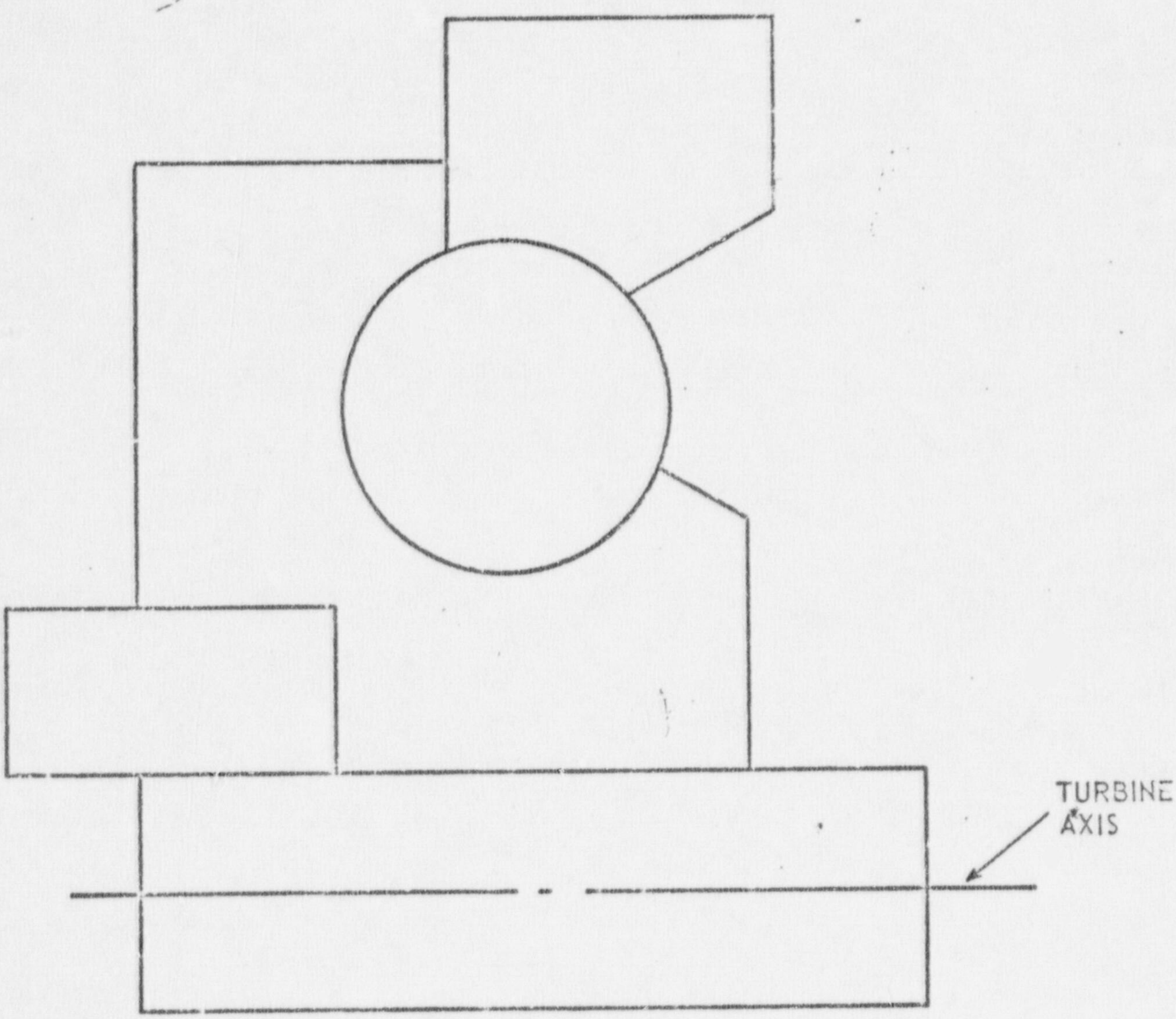
$$P_4 = P_1 \times P_2 \times P_3$$

Missile Damage Probability

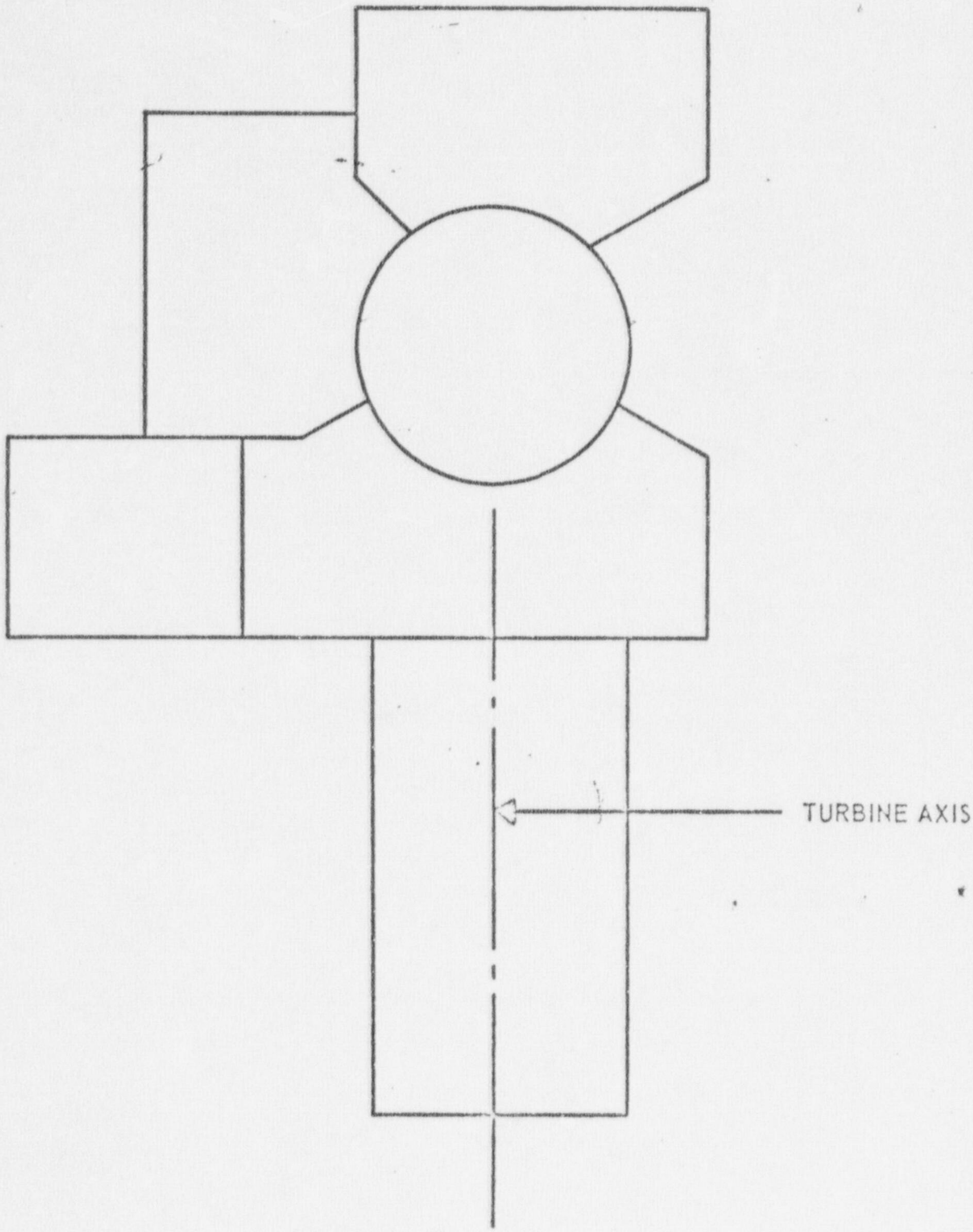
175




DISC EJECTION SITES RELATIVE TO GROUND PLANE




"TANGENTIAL" ORIENTATION




"RADIAL" ORIENTATION

30

# TURBINE MISSILES

ESTIMATION OF DAMAGE POTENTIAL

$P_3$ = PROBABILITY OF A GIVEN MISSILE INCURRING UNACCEPTABLE DAMAGE ASSUMING THAT GENERATION AND STRIKE HAVE OCCURRED.

30

## DEFINITION OF $P_3$

$M_i$ = DEFINED MISSILE

$V_{MIN_i}$ = MINIMUM POSTULATED VELOCITY

$V_{MAX_i}$ = MAXIMUM POSTULATED VELOCITY

$V_{CR_i}$ = MISSILE VELOCITY REQUIRED TO REACH SAFETY SYSTEM

$P_3$ = 0 FOR $V_{CR_i} > V_{MAX_i}$

$P_3$ = 1 FOR $V_{CR_i} < V_{MIN_i}$

$0 < P_3 < 1$ FOR $V_{MIN_i} < V_{CR} < V_{MAX_i}$

## PERFORATION THICKNESSES OF DESTRUCTIVE OVERSPEED TURBINE MISSILES

| MISSILE FRAGMENT[1] | MISSILE WEIGHT(lbs) | IMPACT AREA ($ft^2$) | MISSILE VELOCITY (fps) | PENETRATION[2] DEPTH(Ft) | PERFORATION[2] THICKNESS(Ft) |
|---|---|---|---|---|---|
| I-a | 2000 | 1.50 | 510 | 1.6 | 3.2 |
| II-a | 4000 | 2.19 | 520 | 2.3 | 4.5 |
| III-a | 8200 | 5.07 | 650 | 2.7 | 5.3 |

## PERFORATION THICKNESSES OF TYPICAL TORNADO MISSILES[3]

| | | | | | |
|---|---|---|---|---|---|
| WOODEN PLANK | 125 | 0.333 | 280 | 0.18 | 0.35 |
| SOLID STEEL ROD | 8 | 0.005 | 200 | 0.42 | 0.83 |
| 6" SCH. 40 PIPE | 285 | 0.239 | 155 | 0.19 | 0.38 |
| COMPACT AUTO | 2000 | 7.35 | 180 | 0.06 | 0.12 |

NOTE: $f_c'$ = 3000 PSI

$K$ = 0.0035 = PENETRATION COEFFICIENT

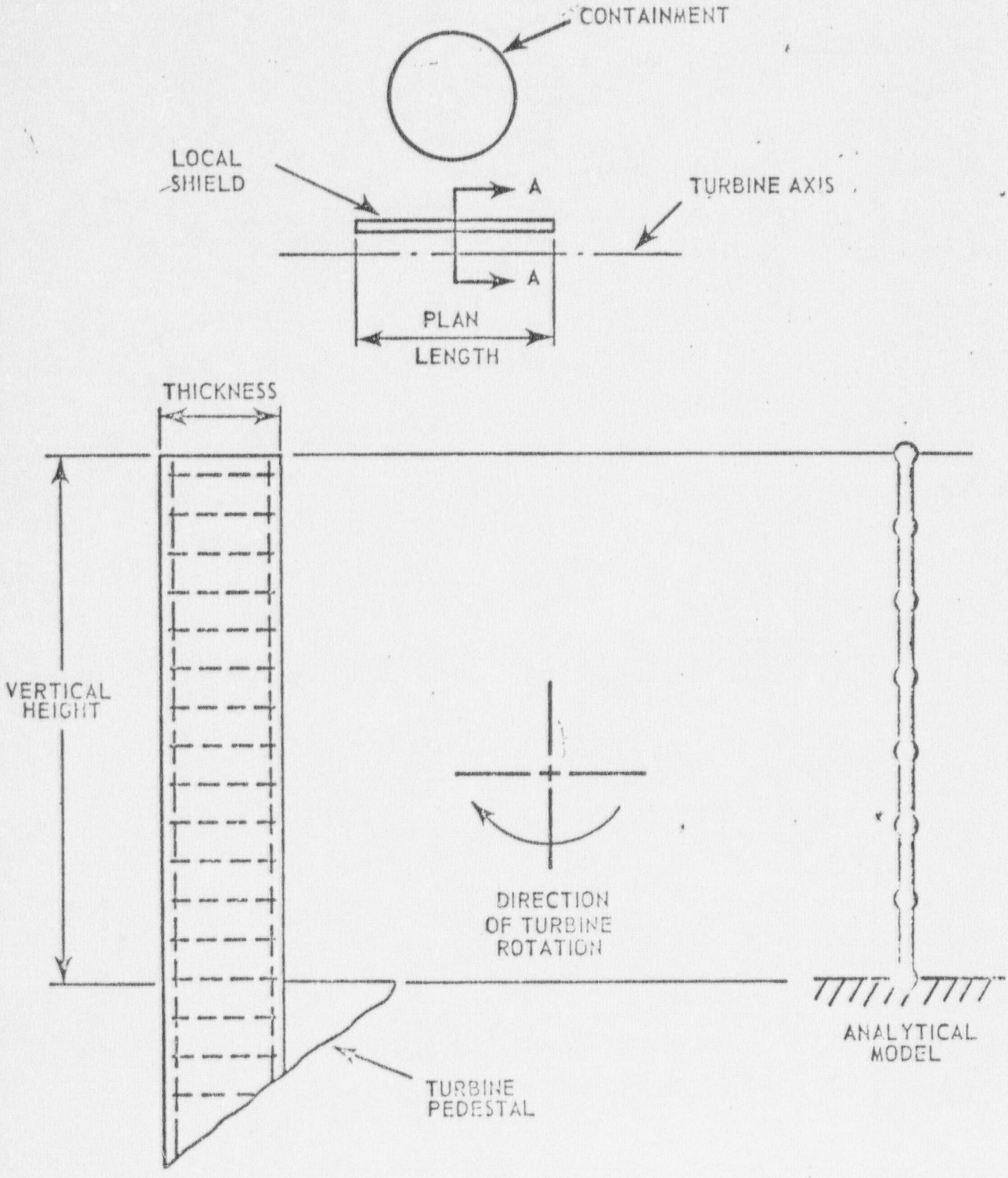
(1) G.E. Memo Report , "Hypothetical Turbine Missile Data, 43-Inch Last Stage Bucket Units", 3/15/73

(2) "Design of Protective Structures (A New Concept of Structural Behavior)", NAVDOCKS p. 51, Annual Meeting of ASCE, October 1950, A. AMIRIKIAN

(3) Topical Report - "Design Parameters for Tornado Generated Missiles", Report No. GAI-TR-102

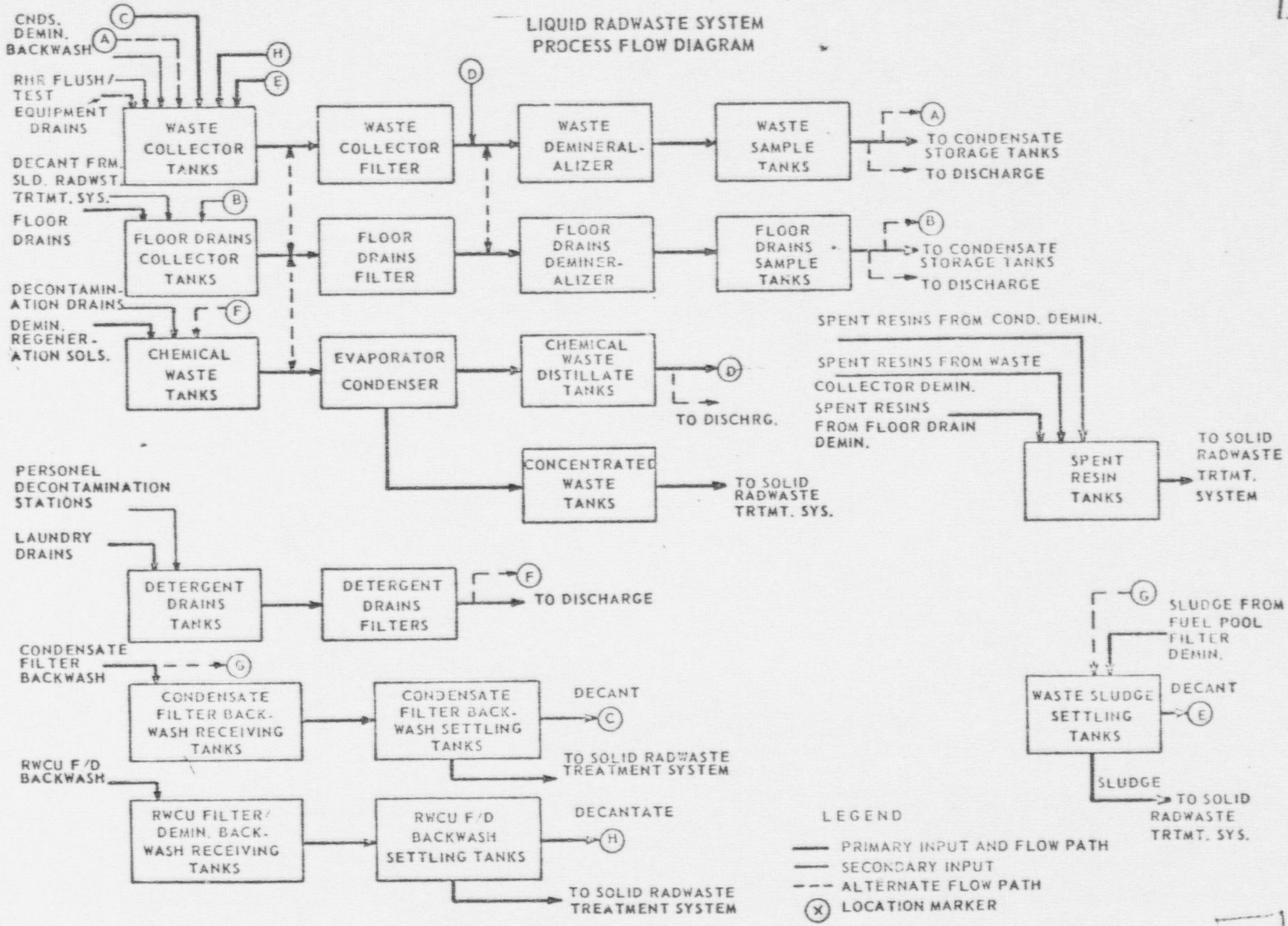
## 120% DESIGN OVERSPEED TURBINE MISSILES

| MISSILE FRAGMENT | MISSILE WEIGHT(lbs) | IMPACT AREA($ft^2$) | MISSILE VELOCITY (fps) | PENETRATION DEPTH(ft) | PERFORATION THICKNESS(ft) |
|---|---|---|---|---|---|
| III-a | 8200 | 5.07 | 420 | 1.47 | 2.95 |
| III-b | 4100 | 3.91 | 530 | 1.33 | 2.67 |
| III-c | 1400 | 1.67 | 610 | 1.28 | 2.57 |
| III-d | 200 | .44 | 800 | .94 | 1.89 |

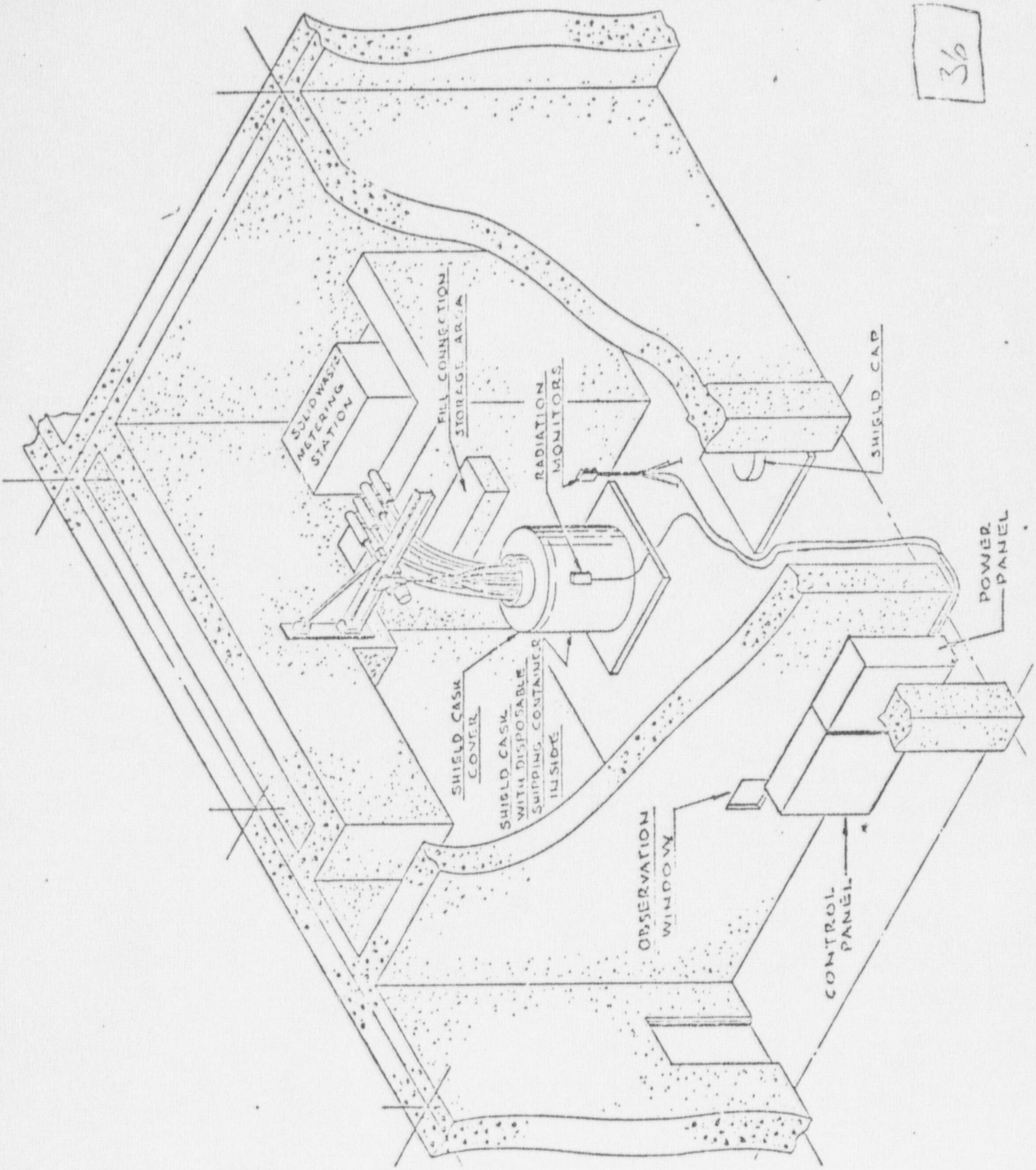



LOCAL SHIELD CONCEPT

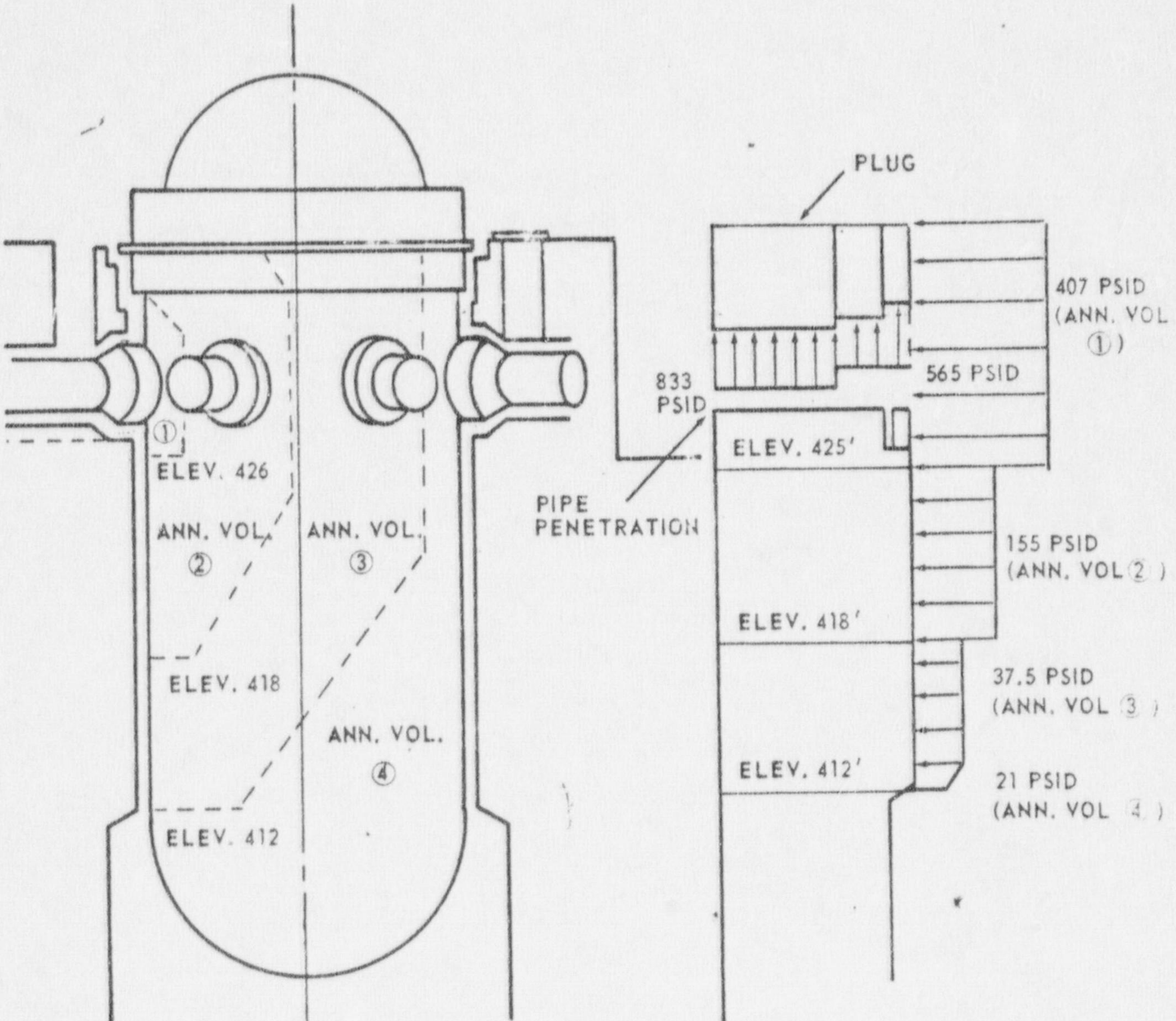
## LIQUID RADWASTE SYSTEM PROCESS FLOW DIAGRAM




35

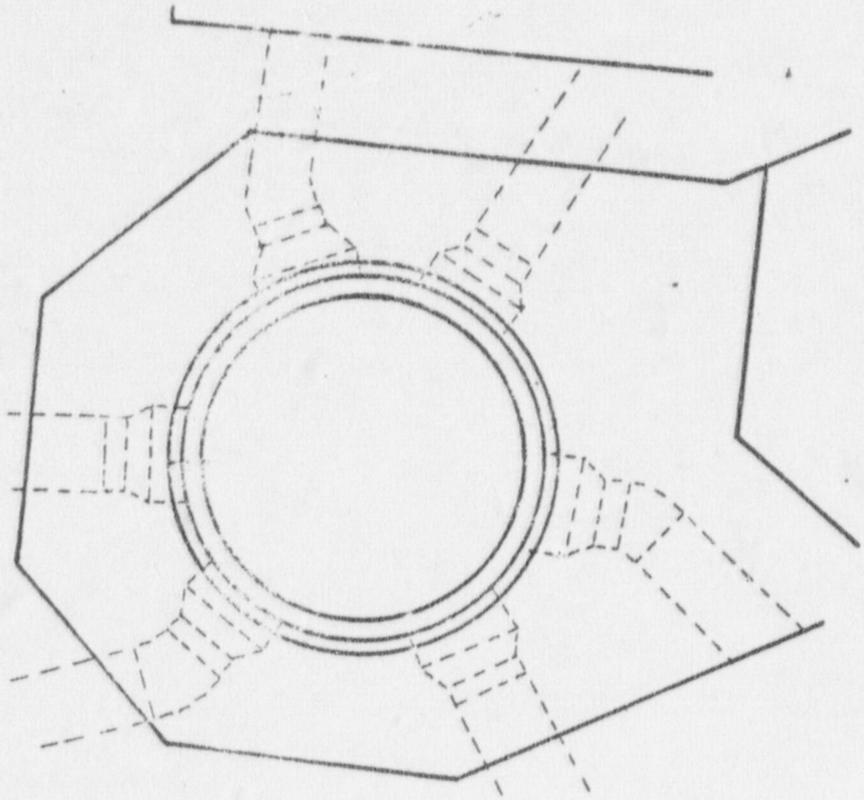


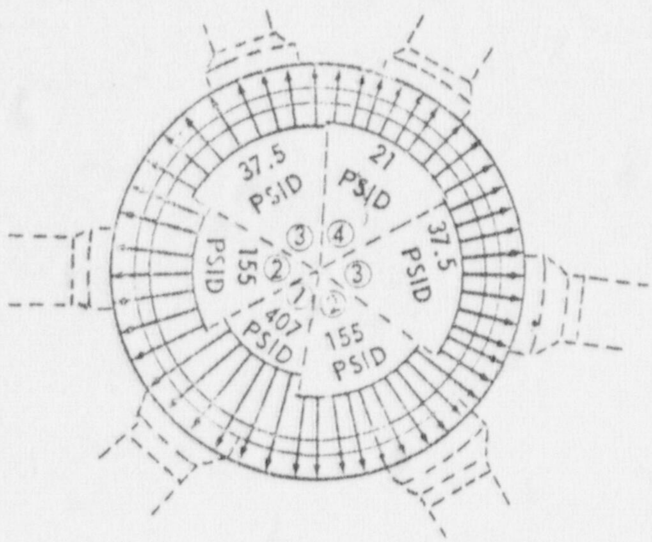

SOLID WASTE PACKAGING OPERATION



PLUG
407 PSID
(ANN. VOL ①)
565 PSID
833
PSID
ELEV. 425′
PIPE
PENETRATION
155 PSID
(ANN. VOL ②)
ELEV. 418′
37.5 PSID
(ANN. VOL ③)
ELEV. 412′
21 PSID
(ANN. VOL ④)
①
ELEV. 426
ANN. VOL.
②
ANN. VOL.
③
ELEV. 418
ANN. VOL.
④
ELEV. 412

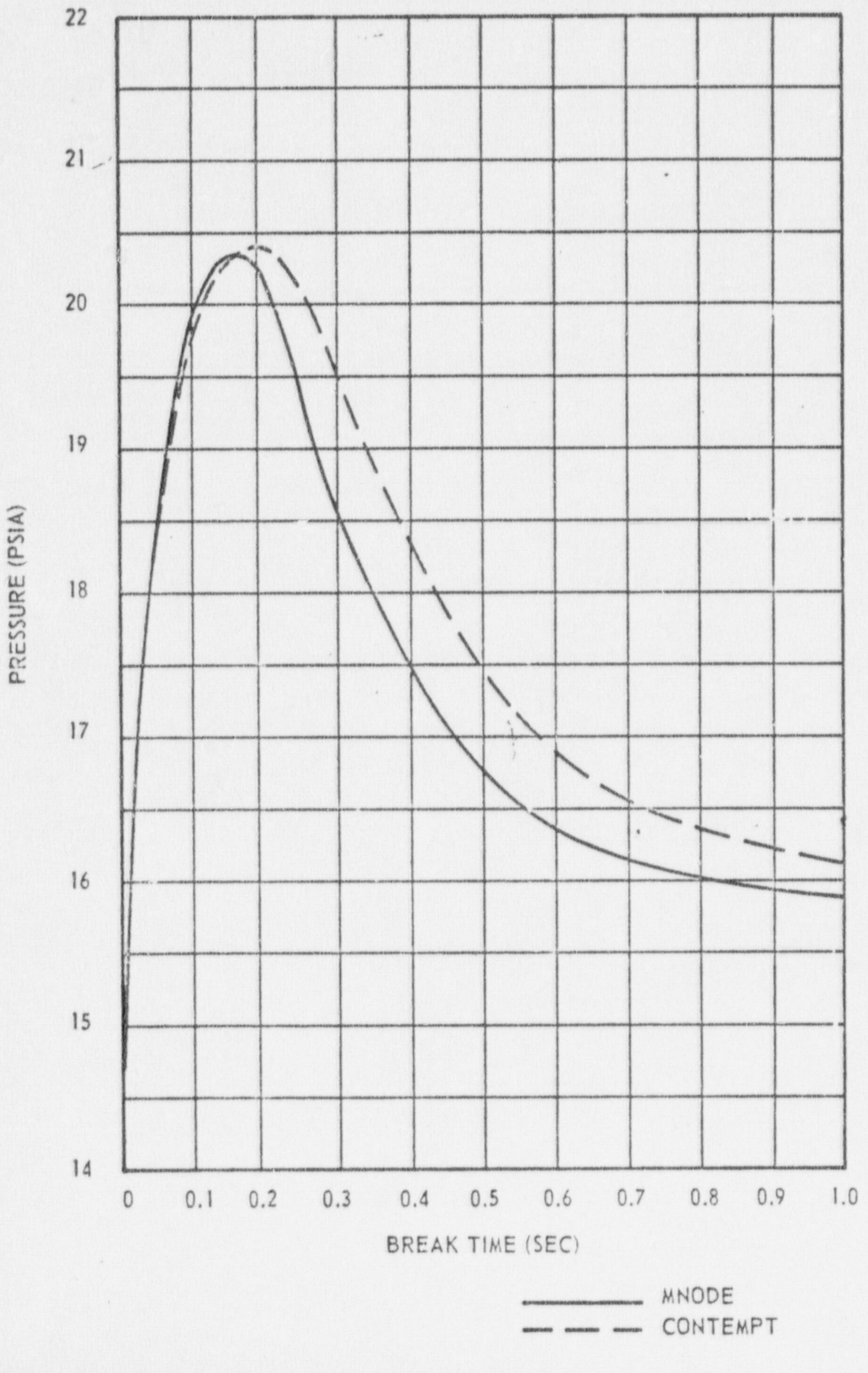

37

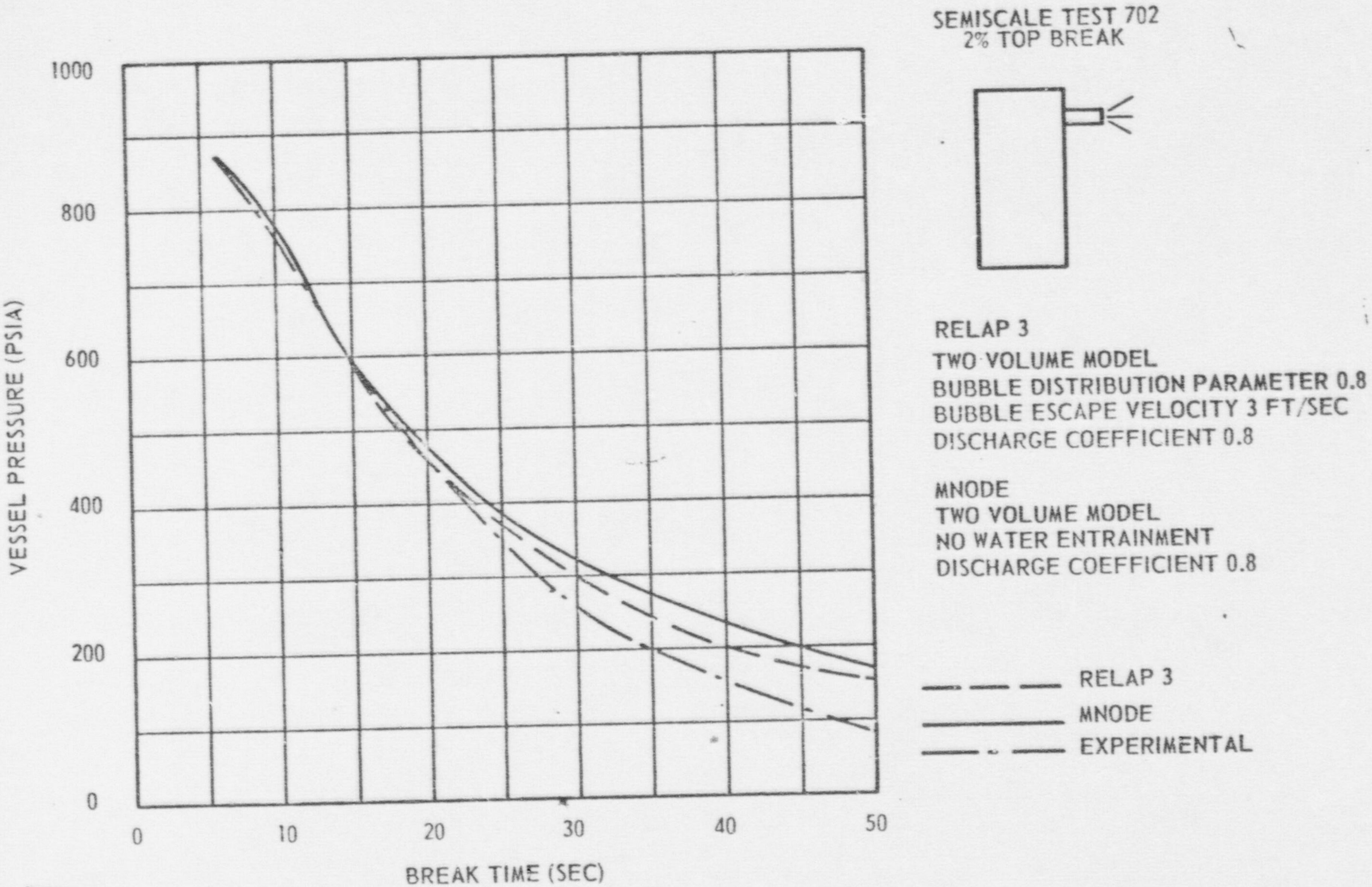


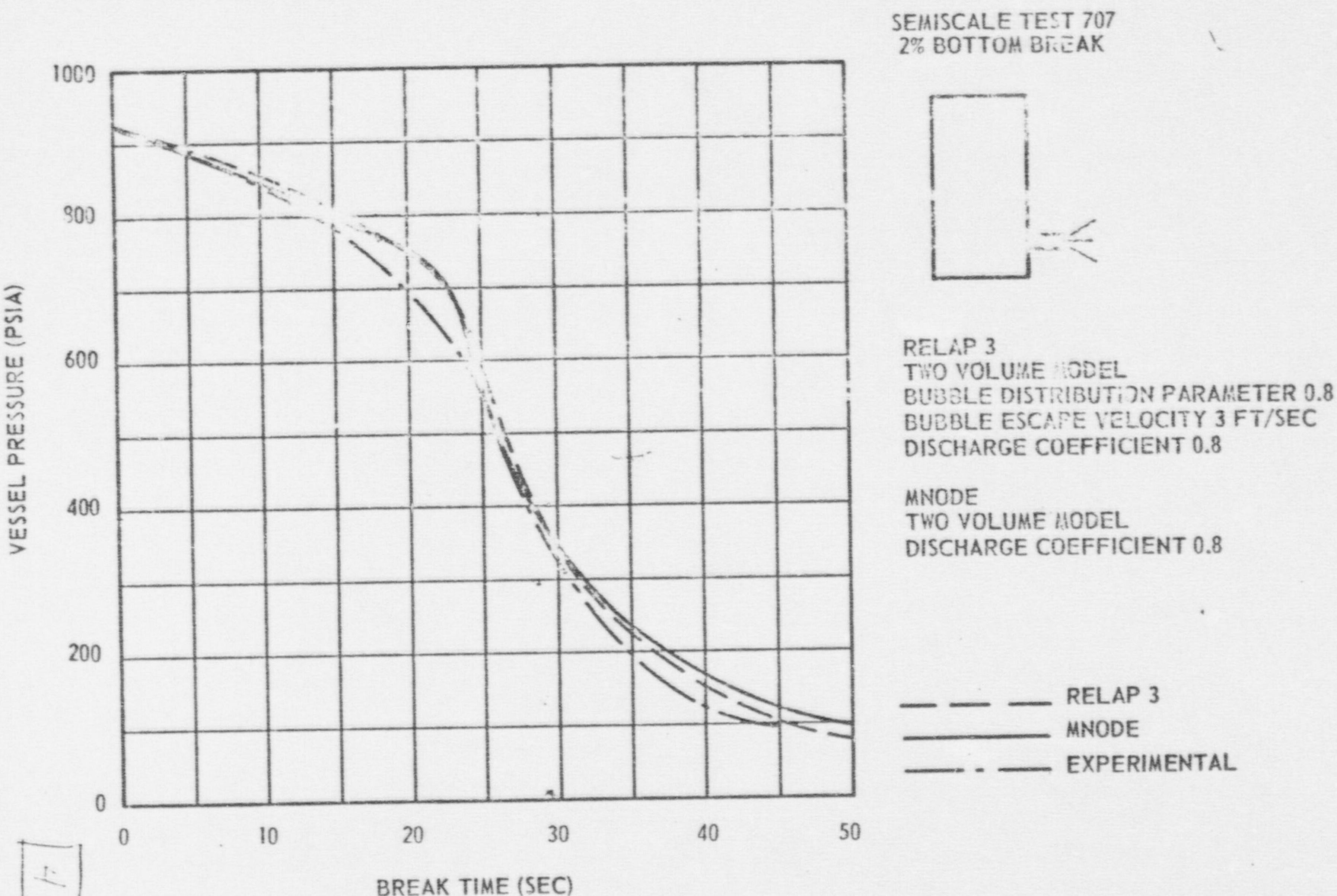


37.5
PSID
21
PSID
37.5
PSID
155
PSID
407
PSID
155
PSID
3
4
2
3
1
2

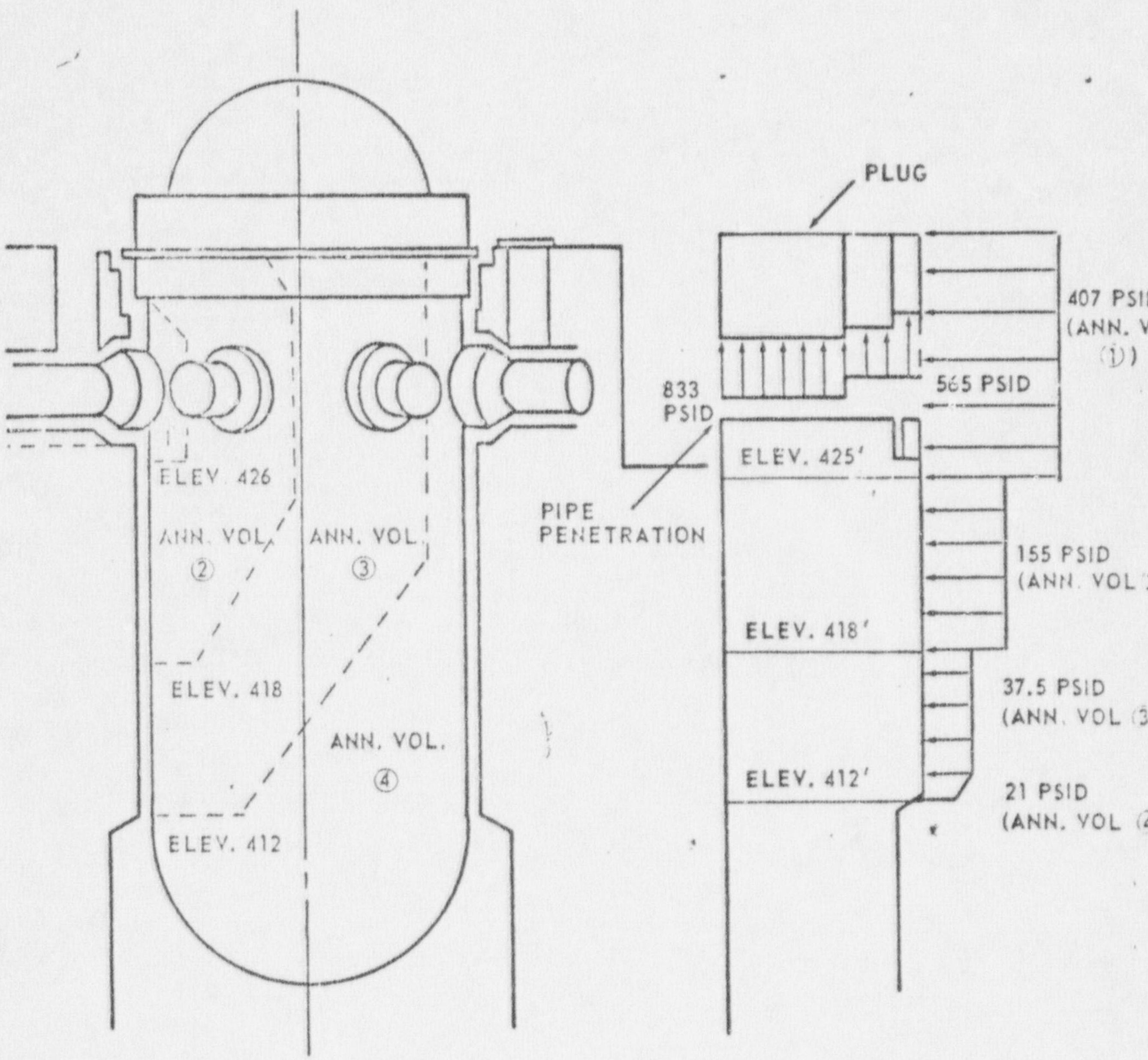
TWO VOLUME MODEL
DISCHARGE COEFFICIENT 0.6

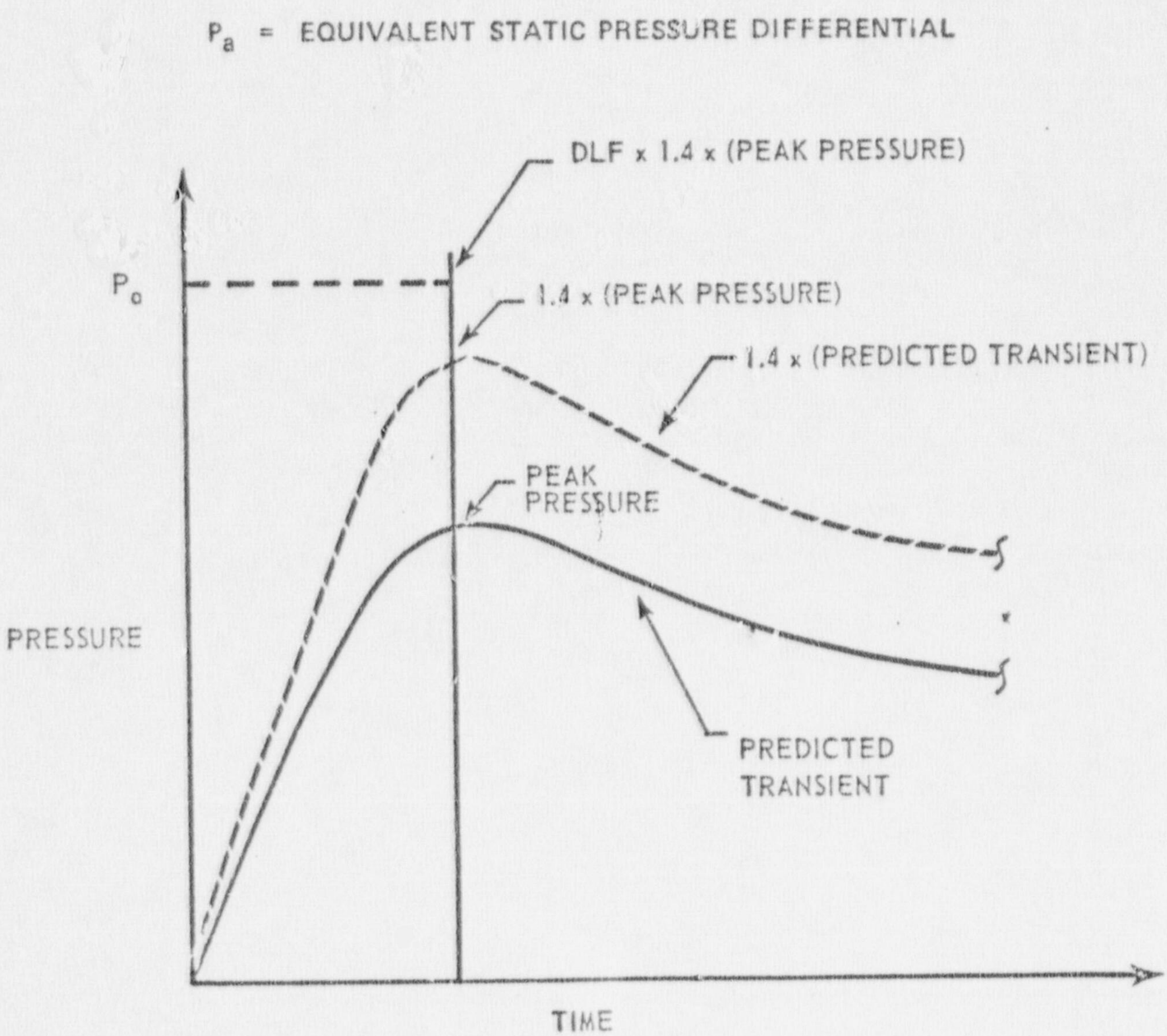



SEMISCALE TEST 702
2% TOP BREAK
VESSEL PRESSURE (PSIA)
1000
800
600
400
200
0
0
10
20
30
40
50
BREAK TIME (SEC)
RELAP 3
TWO VOLUME MODEL
BUBBLE DISTRIBUTION PARAMETER 0.8
BUBBLE ESCAPE VELOCITY 3 FT/SEC
DISCHARGE COEFFICIENT 0.8
MNODE
TWO VOLUME MODEL
NO WATER ENTRAINMENT
DISCHARGE COEFFICIENT 0.8
RELAP 3
MNODE
EXPERIMENTAL


SEMISCALE TEST 707
2% BOTTOM BREAK
RELAP 3
TWO VOLUME MODEL
BUBBLE DISTRIBUTION PARAMETER 0.8
BUBBLE ESCAPE VELOCITY 3 FT/SEC
DISCHARGE COEFFICIENT 0.8
MNODE
TWO VOLUME MODEL
DISCHARGE COEFFICIENT 0.8
RELAP 3
MNODE
EXPERIMENTAL
VESSEL PRESSURE (PSIA)
1000
800
600
400
200
0
0
10
20
30
40
50
BREAK TIME (SEC)

17


PLUG
407 PSID
(ANN. VC
(1))
565 PSID
833
PSID
ELEV. 426
ANN. VOL
(2)
ANN. VOL.
(3)
ELEV. 418
ANN. VOL.
(4)
ELEV. 412
PIPE
PENETRATION
ELEV. 425'
155 PSID
(ANN. VOL 2
ELEV. 418'
37.5 PSID
(ANN. VOL (3
ELEV. 412'
21 PSID
(ANN. VOL (2




PRESSURIZATION TRANSIENT

## LOAD COMBINATIONS

$$U = D + L + T_a + R_a + 1.5\ P_a$$

$$U = D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E$$

REFERENCE: "STRUCTURAL DESIGN CRITERIA FOR CATEGORY I STRUCTURES OTHER THAN CONTAINMENT," DOCUMENT (A), STRUCTURAL ENGINEERING BRANCH, DIRECTORATE OF LICENSING, USAEC, REVISION I, JUNE, 1974

G0563

# OFFICIAL USE ONLY

ACRS-1154A

ATTENDEES OF THE
SAFETY FEATURES PROVIDED BY ARCHITECT
ENGINEERS (GILBERT ASSOCIATES INC.) SUBCOMMITTEE
WASHINGTON, D. C.
AUGUST 23, 1974

| ACRS | GILBERT ASSOCIATES INC. |
|---|---|
| H. S. Isbin, Subcommittee Chairman | H. Areur |
| M. Bender | K. R. Broome |
| S. Lawroski | D. K. Croneberger |
| *G. R. Quittschreiber, Staff | A. Gannon |
| | S. D. Goodman |
| *Designated Federal Employee | E. R. Hottenstein |
| | G. M. Kowal |
| | R. Mathys |
| | T. M. McMahon |
| | W. E. Meek |
| | F. Moreadith |
| | L. B. Myers |
| | H. R. Porter |
| | D. A. Raquet |
| | W. B. Shields |
| | D. P. White |

PUBLIC-AT-LARGE

| Name | Affiliation |
|---|---|
| E. Commell | FLUOR PIONEER, INC. |
| Bruce Deering | BROWN & ROOT, INC. |
| Lawrence Doones | |
| Orest A. Hrynewych | SARGENT & LUNDY |
| M. S. Kumar | GIBBS & HILL, INC. |
| D. C. Marlwyer | WESTINGHOUSE |
| T. Sahlin | FLUOR PIONEER, INC. |
| Edward J. Scheetz | WESTINGHOUSE |
| John B. Ross | COMMONWEALTH ASSOC. |
| Janet Utbaci | |

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3/12/75

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ACRS-1158A

April 28, 1975

Meeting Date: 9/11/74
Date Issued MAY 7 197

GENERAL ELECTRIC SUBCOMMITTEE MEETING ON
GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT (GESSAR)
WASHINGTON, D.C., SEPTEMBER 11, 1974

The General Electric Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) met in Room 1046 at 1717 H Street, N.W. in Washington, D.C. on September 11, 1974 as announced in the Federal Register on August 23, 1974. The purpose of the meeting was to further review the General Electric Standard Safety Analysis Report (GESSAR).

Attendees

ACRS

H. Isbin, Subcommittee Chairman
L. Fox, ACRS Member
H. Monson, ACRS Member
J. McKinley, ACRS Staff

AEC Regulatory Staff

V. A. Moore
J. F. Stolz
D. Crutchfield
L. Shothin
T. M. Novak

General Electric Company

K. M. Ketchel
P. W. Marriott
A. E. Rogers
B. P. Grim
A. J. James
J. L. Embley
I. F. Stuart
W. D. Gilbert
J. F. Quirk

Public

R. E. Schaffstoll - B&W
James A. Domer - TVA
Ira W. Merritt - TVA/ODEC
J. M. Gibbons - Bechtel
Charles R. Wienke - Bechtel
A. E. Toombs - S&W
R. R. Brems - GAI
K. L. Howard - VE&C

Executive Session (Closed)

Dr. Isbin characterized the preceeding Subcommittee meeting (July 1, 1974) as GE's presentation of an overview of the concept with no details. It did develop the fact that the water depth in the suppression pool had been reduced. The Perry Station applicant has cited Amendment 16 to GESSAR as justification for his reduction in suppression pool depth.

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Another topic that needed to be addressed was that of ECCS analysis. Dr. Isbin noted that GESSAR does not directly include either the General Electric Thermal Analysis Basis (GETAB) or the GEXL correlation.

Dr. Isbin interperted the ACRS position to be that GES AR should resolve some of the major unresolved generic issues such as: ECCS, ATWS, and Mark III containment. Other issues such as prompt relief trip and instrumentation and control also need to be addressed. It did not appear that hydrogen generation and control was an issue for GESSAR.

It was noted that, hopefully, once the ACRS has reviewed and approved GESSAR it will not have to go back and review the details of a plant that references GESSAR. It was also noted that the ACRS has always felt free to reevaluate anything it wanted to and there was nothing to preclude a re-review of a previously accepted position.

A number of other topics were identified for possible discussion, including; leak detection and location, industrial security, reliability analysis, and recirculation pump coastdown.

No Regulatory Staff report had yet been received on GESSAR.

Meeting with the General Electric Company and the AEC Regulatory Staff (Open)

Dr. Isbin began the open portion of the meeting by making an introductory statement regarding the conduct of the meeting in accordance with the Federal Advisory Committee Act. He noted that no statements from the public had been received and no requests had been made to make oral statements. He briefly reviewed the July 1, 1974 meeting of the Subcommittee.

Mr. Stuart, Manager of Safety and Licensing for GE's Nuclear Energy Division, expressed GE's appreciation for the opportunity to present its case to the Subcommittee and urged that GESSAR be included on the ACRS's November agenda. Mr. Stuart distributed an agenda of topics that GE would like to cover at this meeting (see Figure 1).

Mark III Containment

Mr. James (GE) reviewed the test program that had been conducted to date to confirm the performance of the Mark III containment system. He reviewed the tests that had been conducted in the large-scale pressure test facility at San Jose. He described the test facility and three series of tests that have been run to date. The first series involved one, two, and three vent steam blowdowns, the second series involved air blowdowns, while the third series involved three vent, one third scale steam blowdowns.

# AGENDA FOR ACRS SUBCOMMITTEE
# REVIEW OF GESSAR
# SEPTEMBER 11, 1974

1. MARK III TEST PROGRAM AND DESIGN
2. 10 CFR 50 APPENDIX K RESULTS
3. GETAB
4. CONTROL AND INSTRUMENTATION
5. ATWS
6. ITEMS BEING RESOLVED WITH STAFF
7. OTHER ACRS TOPICS

FIGURE 1

9/11/74

Mr. James presented a table showing the relative ratios (scaling) of various parameters of the test rig (see Figure 2). He noted that the air volume, pool volume, and pool surface were not full scale. Some of the scaling ratios can be varied by varying the vent size and number, and the pool geometry by means of baffles.

The first test series was to confirm that the Mark III system would work and would condense steam as had been predicted. The results of this series appeared to confirm the conservatism of the analytical model. The test series confirmed that the vent clearing model is what controls the calculated peak pressure in the drywell and the tests showed that the vents actually cleared faster than calculated.

The second series of tests was to evaluate the impact loadings of the water splashing from the pool surface against the undersides of structures located above the pool. Pool swell occurs as the air in the drywell is pushed into the pool, before the bubbles break the surface. Impact targets were located above the pool surface. The experimenters were able to achieve the air flow rates that would be representatives of the Mark III system. These resulted in plots of water surface profiles as a function of time (see Figure 3). When the ligament of water above the bubble decreases to 2 or 3 feet it becomes unstable and the bubble breaks through in many places creating a froth that is thrown into the air. Breakthrough occurs at an elevation of roughly 1.2 times the vent submergence. Mr. James claimed that there was nothing unexpected in the results of this series of tests.

Rather high loads were measured by the impact targets. At some locations average surface pressures of up to 115 psi were measured for very short durations (seven milliseconds).

It was also found that about twenty feet above the pool surface the mixture was primarily air with some water entrainment. This was useful in evaluating the flow past the constriction of the hydraulic control unit (HCU) floor. This is important because a restriction of ait flow past the HCU floor by the two phase mixture could result in a pressure buildup beneath the floor and excessive structural loads. Froth pool swell can be expected to a height of over thirty feet.

No attempt was made during this series to make a sensitivity comparison between initial reactor vessel pressure and impact loadings on structures above the pool.

The third series of tests was run after the test facility had been extensively modified to produce a 1/3 scale model of the weir annulus, vents, and suppression pool volume and surface area, and drywell volume. The objective of this test series was to measure the pressure rise in the space above the

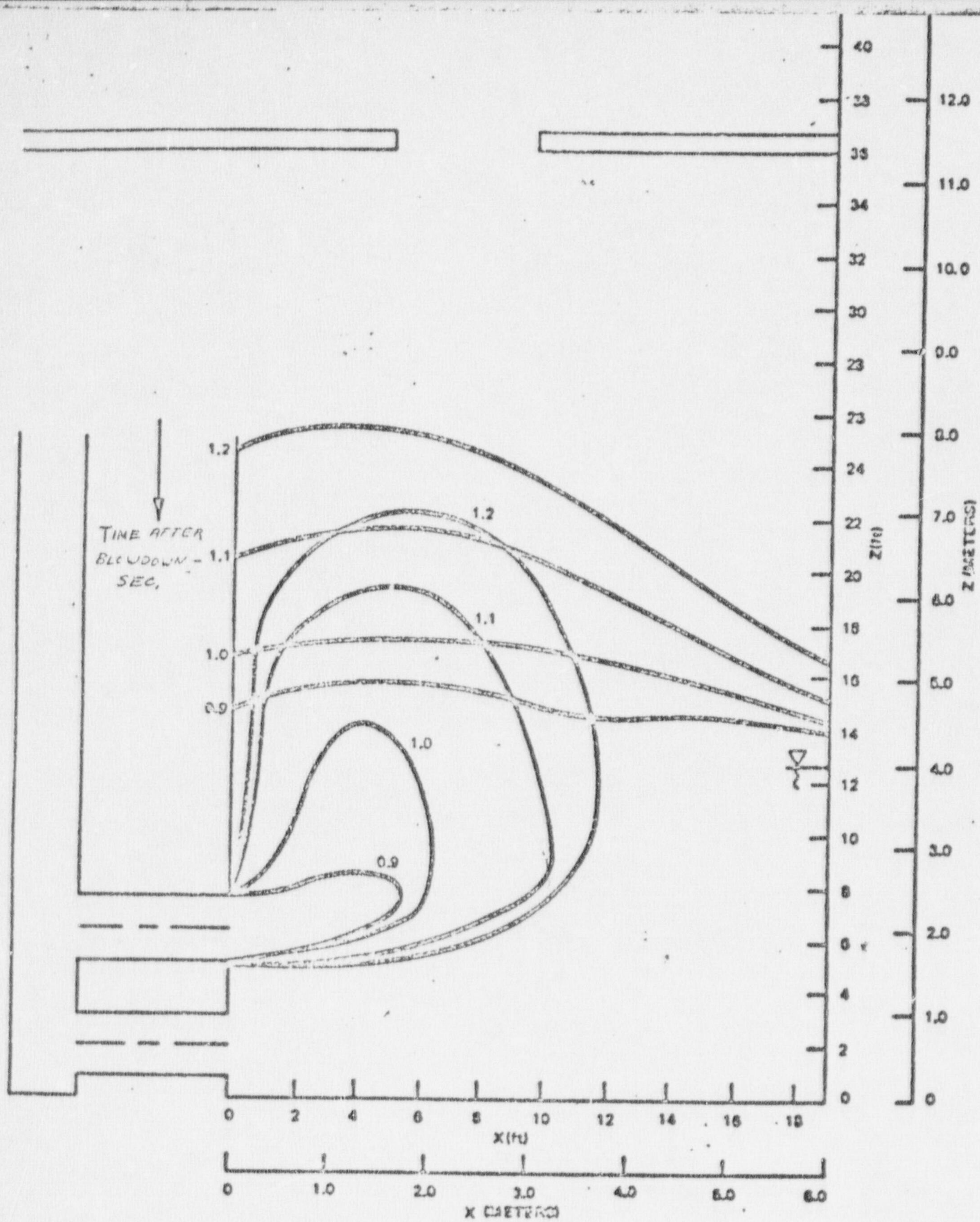
# FULL SCALE MARK III TESTS

## SCALING

| | RIG | FULL SCALE | RATIO |
|---|---|---|---|
| VESSEL VOLUME, $FT^3$ | 160 | 22,000 | ~150 |
| DRYWELL VOLUME, $FT^3$ | 2065 | 200,000 | ~120 |
| POOL VOLUME, $FT^3$ | VARIABLE (3750 MAX.) | 150,000 | ~40 |
| POOL AREA | | | 40 |
| SUPPRESSION CHAMBER FREE VOLUME, $FT^3$ | 10,000 | $1.4 \times 10^6$ | 140 |
| BREAK AREA, $FT^2$ | VARIABLE (.1 MAX.) | 4.5 | 45 |
| VENT AREA, $FT^2$ | VARIABLE (12.0 MAX.) | 530 | 40 |
| BREAK AREA/VENT AREA | .008 | .008 | 1 |
| SUBMERGENCE | ~10 | ~10 | 1 |

A.J.J. 6/73

FIGURE 2




Pool Swell and Bubble Location Map

FIGURE 3

suppression pool due to the restriction of air flow by the frothy mixture passing the HCU floor. In addition, data were obtained on pool swell, froth impingement, vent clearing, and drywell pressure response. It was found that for Mark III design conditions, with a 25% clear passage past the HCU floor, the peak differential pressure across the floor is about 6 psi, which corresponds with a calculated value of 11 psi.

Mr. James described the next series of 1/3 scale tests which are directed at obtaining confirmatory impact data on various structural steel shapes, pipes, gratings, and electrical penetrations. Planning beyond this series is very preliminary and subject to change but the facility is tentatively scheduled to be returned to the full-scale configuration in early 1975, with three vent tests to be begun in May 1975. Pool temperature tests are scheduled for the second and third quarter of CY 1975. Small steam blowdowns into a large pool are scheduled for the last quarter of 1975 in order to investigate the potential for pool stratification. During the first quarter of 1976 some liquid blowdowns are scheduled to confirm that the steam blowdown is controlling for the containment design. Multi (9) vent tests are also planned but the timing is not clear. Mr. James did not believe that multi-vent tests were necessary to confirm the validity of the pool swell loadings on structures above the pool. The multi-vent tests are intended to confirm that there are no horizontal interactions between vents.

Some questions were raised on the schedule and the priorities of various test series. Mr. James noted that the test facility was booked solid until at least the middle of 1976 and the priorities were established primarily to meet Staff needs for licensing.

Mr. Crutchfield (Regulatory Staff) said that the AEC was very satisfied with the results of the GE test program thus far. He complimented GE on the way it had accommodated the changes requested by the Staff. He pointed out that the Staff itself is auditing both GE and the architect-engineers doing the detailed design of the Mark III. The Staff and the A-Es are trying to develop independent analytical tools to model containment performance.

Mr. James next addressed design changes that have been made in the Mark III system as a result of the test program. These include:

1. Reduction of the top vent submergence from 11 feet to 7 1/2 feet and the addition of a means to add water to the pool later in the transient.

2. The structure known as the reactor water clean-up floor was relocated out of the annular space between the drywell and containment.

3. Several small structures were lowered to the pool surface to eliminate the impact loads.

4. The bridge between the containment equipment penetration and the drywell wall was made removable and will be stored on the operating floor.

5. The remaining structures above the pool will be designed to withstand the impact loads resulting from pool swell.

The pool level was lowere', not only to reduce the impact load on the hydraulic control unit (HCU) floor but also to improve the air flow through the space between that floor and the containment by reducing the amount of liquid in the froth. The improved air flow reduced the differential pressure across the HCU floor and the pressure occurring in the drywell.

One criterion for the Mark III is that the submergence of the top vent should never be less than two feet. ECCS pumps take suction out of the pool and deliver water to various locations where a significant volume of it gets held up and results in a reduction of vent submergence. In order to keep the vents properly submerged, additional water must be supplied to the pool from a Category I supply system taking water from the upper containment pools. The system is designed so that if there is an inadvertent dump of the upper pools into the suppression pool, the water will not overflow the weir wall and will not flood the drywell. The supply system will be redundant and testable and either of the two supply lines will provide adequate makeup flow. The system will be locked closed in the refueling mode of operation. The make-up system will be actuated by a LOCA signal coincident with a low-low water level signal.

Emergency Core Cooling System Calculations

Mr. Marriott (Manager, Emergency Core Cooling Systems Engineering, GE) reviewed the status of GE's reanalysis of ECCS using the criteria and requirements of Appendix K. He noted that the analysis of ECCS performance is covered in Section 6.3 of GESSAR and that Amendment 19 addressed the Appendix K requirements.

Mr. Mariott pointed out that GE met with the Staff many times in developing its new models and formal submittals are being prepared. The draft report has been assigned the number NEDO-20566, a final report is scheduled for the end of 1974. GE believes its model for Appendix K is essentially complete. A few documentary and minor technical issues still need to be cleaned up, but resolution is expected shortly.

Mr. Mariott presented the following table comparing the acceptance criteria with the calculated values for BWR/6.

BWR/6 238-732

Conformance to 10 CFR 50.46

| CRITERION | LIMIT | BWR/6 |
|---|---|---|
| Peak Cladding Temp | 2200F | 1550F * |
| Peak Local Oxidation | 17% | 0.5% |
| Core-Wide $H_2$ Generation | 1% | $<$ 0.02% |
| Coolable Geometry | YES | YES |
| Long-Term Cooling | YES | YES |

* Flat local power distribution, gap conductance at lowest value as function of exposure.

One procedural change was mde in the analysis that resulted in an increase in the calculated peak clad temperature, this was the introduction of GEGAP-3 a new gap conductance model. In addition, all pins are assumed to be operating at the design linear heat generation rate of 13.4 kw/ft., this assumption results in a 100-150°F increase in the calculated peak clad temperature.

GE has determined that even under the worst fuel duty conditions and highest internal gas pressure, the perforation temperature for the 8x8 fuel rods is considerably in excess of 2200°F and thus no perforations or internal clad oxidation are expected.

The final ECCS acceptance criteria have not produced a dramatic difference in BWR evaluations.

In the analysis of the LOCA, the recirculation pumps are assumed to coast down with a time constant of about five seconds (that is not a precise definition of the coast down characteristics but is sufficiently close).

The Staff has recently asked GE to evaluate the consequences of a closure of the flow control valve in the unbroken loop of BWR/6 during a DBA. In prior analyses the pump in the unbroken loop was assumed to coast down and provide some forced circulation during the initial part of the blowdown. The GE analysis shows this situation to be no worse than others studied. The study results had not yet been transmitted to the Staff.

For pipe breaks 2 inches in diameter or smaller, the reactor vessel will not be fully flooded after 10 minutes, diversion of one of the LPCI pumps to the containment spray function at this time will delay the complete filling of the reactor vessel. This delay results in a rather small increase in an already small calculated peak clad temperature.

Noting the rather low calculated peak clad temperature for the BWR/6, Dr. Isbin asked what GE intended to do with regard to the ACRS's expressed desire for improved ECCSs. Mr. Marriott thought GE would strive for a better understanding of the phenomena relating to the LOCA and ECCS effectiveness and to further assure that the margins of conservatism, believed to be present, continue to exist. Mr. Stuart said that GE was also seeking standardization in ECCS as well as in the rest of the plant. GE was not prepared to address reliability criteria; Mr. Stuart made a major distinction between reliability and the single failure criterion. Dr. Isbin expressed the Committee's continuing interest in reliability and suggested that it could be a topic for detailed discussion at a future meeting. Mr. Stuart pointed out that GE, too, has an interest in reliability and is participating in a program to collect data on systems and components.

Thermal Analysis Basis

Mr. Rogers (Manager, Nuclear Steam Supply Thermal Hydraulics, Nuclear Energy Division, GE) summarized the General Electric Thermal Analysis Basis (GETAB) as it applies to BWR/6. The design basis for GETAB is that transients caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in monitoring the core operating state, more than 99.9% of the fuel rods would be expected to avoid boiling transition Mr. Rogers explained in detail how the operating limit of 1.21 for the minimum critical power ratio was established based on ATLAS test data and multiple uncertainties. He compared the ATLAS test facility parameters with those of BWR/6 to show the validity of the ATLAS results. A study of the ATLAS data aided in the development of the GEXL correlation to predict critica core parameters. GE believes GEXL can predict core behavior to within about 3%.

Mr. Novak (DRL) said that the Staff was completing its review of GETAB and would issue a report of its findings in the next few weeks.

Mr. Rogers pointed out that the operating limit chosen is to assure fuel integrity (99.9% of the rods won't fail) for the most severe design transient at the most adverse time in core life. GE does not believe fuel rods will fail immediately upon the departure from nucleate boiling and this is factored into the correlation.

Anticipated Transients Without Scram (ATWS)

Mr. Stuart expressed GE's concern that its present reactor protection system is being considered inadequately reliable and that the AEC believes there is need of some back up system. The reliability requirements for the back-up system are not defined. He thought that GE had demonstrated and documented that its system is adequately reliable. He said that some of GE's customers are urging GE to resist additional requirements regarding ATWS for BWRs.

A Committee member pointed out that neither shutdown system alone might be shown to have adequate reliability to prevent an ATWS but two together might have sufficiently high reliability.

Mr. Embley (GE) reviewed the history of the ATWS concern which was first mentioned in the ACRS report on Hatch, Unit 1 in May 1969. GE submitted a topical report on common mode failures in July 1970 and ATWS in March 1971. The ATWS report was supplemented in April 1973. GE believes that a failure to scram is improbable but if it should occur, tripping the recirculation pumps and manually initiating liquid boron solution injection will control the situation.

The AEC position was published in September 1973 in WASH 1270. The Staff criteria were much more restrictive than those established by GE. Mr. Embley presented a comparison of some of the major differences in criteria (see Figure 4). GE analyzed its designs and found that it could meet the requirements of WASH-1270 and that the temperature of the suppression pool water was the limiting parameter. In order to meet WASH-1270, certain hardware changes are required; including automatic recirculation pump trip on high reactor pressure or low reactor water level, automatic boron injection automatic start of RCIC and HPCS, and trip of the feedwater pumps. A topical report on this topic is scheduled to be issued in October 1974.

The Staff has had Brookhaven National Laboratory make an independent evaluation of an ATWS in a BWR. The results of this study tend to confirm GE's conclusions.

Regulatory Staff Review

Mr. Crutchfield presented the status of the Regulatory Staff's review of GESSAR. A Safety Evaluation Report (SER) is in preparation. The Staff has reviewed the application through Amendment 18; Amendments 19 and 20 had just been received and may resolve some of the current outstanding issues. At the moment the Staff has about 46 open items on this application. Eighteen of those items are areas in which the Staff has taken a position but GE has not agreed to that position, about 9 or 10 items require verification of GE's position or a date for the submittal of additional information (post-PDA), about 14 items require more information before a PDA

# ATWS - CRITERIA COMPARISON

| AEC (WASH 1270) | G.E. (NEDO 10349) |
|---|---|
| 10 CFR 100 | 10 CFR 100 |
| EMERGENCY STRESS NOT EXCEEDED IN PRIMARY SYSTEM | PRESSURE IN VESSEL LESS THAN STRUCTURAL FAILURE POINT |
| CONTAINMENT PRESSURE LESS THAN DESIGN | CONTAINMENT PRESSURE LESS THAN MEMBRANE YIELD LIMIT |
| NO SIGNIFICANT CLAD DEGRADATION | 17% CLAD OXIDATION |
| NO SIGNIFICANT FUEL MELTING | MAX FUEL ENTHALPY 280 CAL/GM |

FIGURE 4

can be issued, and 6 items on which GE has submitted information but have not been completely reviewed by the Staff. The Staff has slipped the expected issue date of its SER one month, to October 9, 1974.

Mr. Crutchfield briefly summarized the open items on GESSAR (see Attachment A) He noted that GE has indicated that a better safety/relief valve design is forthcoming.

The instrumentation and control systems are completely redesigned and the Staff review will be based on design criteria and functional diagrams rather than detailed review of detailed plant drawings.

GE responded to selected items including drywell testing, free water in solid waste, automatic containment spray actuation, and RHR operation with a single failure.

The Regulatory Staff intends to require structural proof testing and leak testing at high pressure for the drywell. GE pointed out that this was a primarily concrete structure with walls about five feet thick. The drywell is totally enclosed in the containment and surrounds the reactor vessel. The drywell is not intended to be a leak tight barrier. To perform a high pressure test on the drywell will require leak tight closures over the 120 vents into the suppression pool while a low pressure (2-3 psi) could be done without closing vents but with having water in the pool. GE had appealed the Staff requirement for a high pressure test and the appeal had been rejected. GE estimates the cost of sush a test to be $4 million to $8 million due to the construction delays it would cause (3 to 6 weeks). GE conceded that the tests could be done but pointed out that the costs are rather high and can affect the marketability of the Mark III containment.

With regard to assuring that there is no free water in the solid waste, GE described the proposed method of solidifying the wastes from the waste concentrator. The products from the concentrator are placed in steel barrels to which an appropriate amount of cement is added and mixed with an internal paddle. The cement absorbs any free water. The paddle is left in place in order to avoid voids and the barrel is sealed. GE believes this will assure no free water in the barrels but also admits there is no instrument currently available to detect free water in the barrels if it was present.

With regard to the Staff requirement for automatic actuation of the containment spray, GE pointed out that, except for certain transients associated with the hydrogen system, the Mark III does not need a containment spray system. Even if there is significant bypassing of the drywell, the containment sprays are not needed for about 10 minutes. Over the long term, after a LOCA, the containment pressure will increase due to the increased temperature of the suppression pool water. GE's basic argument was that there was plenty of time for considered manual operation and automatic operation was not necessary.

Mr. Quirk (GE) discussed the combination of loading conditions and the stress limits allowed for each combination for piping and other system components and equipment. GE is designing its systems to ASME code requirements and to Regulatory Guide 1.48 except for upset plant conditions plus a combination of an abnormal operational transient and a seismic event one-half of the SSE. The analysis for this combination has been made but the allowable stress limit is higher than the Staff allowable. GE objects to the use of two upset events simultaneously and believes the probability of that occurrence is no greater than that of an emergency event and the allowable stresses should be comparable. GE is exploring a possible compromise position that may resolve this issue.

The second item that Mr. Quirk discussed was the use of the RHR system for shutdown cooling and the single failure criterion. The Staff position is that the RHR system must be able to bring the reactor to a cold shutdown condition assuming a single failure in that system. GE has argued that, although a single failure could prevent the RHR system from achieving cold shutdown, there are functionally redundant ways of achieving that goal. Normally, after the reactor is cooled to about 340°F by venting steam to the main condenser, the RHR system is lined up to take suction on one of the recirculation lines, pump water through a heat exchanger and return it to the reactor via the feedwater lines. If, for some reason, the valves to the recirculation loop failed to open, steam could continue to be vented to the main condenser and system temperature and pressure reduced to the point where personnel could enter the containment and repair the valves. Alternate the RCIC turbine could be run and one RHR heat exchanger could be used as a condenser for reactor steam and the condensate pumped back to the reactor.

GE took issue with recently released design criteria for the radwaste system. GE had proposed a system based on Regulatory Guides 1.26 and 1.29 but the proposal was rejected by the Staff on the bases of recently proposed AEC criteria. The new criteria are substantially more restrictive than GE or industry standards. Currently the industry is designing the radwaste buildings as Category I structures and designing them such that the lower portions are a catch basin for any liquid leakage resulting from seismic damage to the liquid waste systems.

The Staff wants the gaseous waste storage tanks (charcoal filled) designed to a quality better than Quality Group D (Group D augmented). In the dose evaluation of failure of the storage tanks, the Staff assumes that all the gas on the charcoal escapes. GE complained that it had not seen the Staff's analytical methods.

Mr. Gilbert presented a comparison of the Staff's assumptions with those of GE and what was reasonably expected (see Figure 5). He went on to tabulate the doses at 300 and 500 meters, calculated for various release assumptions (see Figure 6). Mr. Gilbert interpreted the Staff's requirements to be; take the waste gas system out of the turbine building and put it in a separate seismic Catetory I building. GE believes this is an unnecessary expense. Mr. Gilbert objected strenuously to the quality requirements the Staff is proposing to impose on the radwaste systems.

Instrumentation and Control

Mr. Grim (GE) discussed the nuclear system protection system and the rod control and position system. The latter includes rod pattern control and rod position information.

The functions incorporated into the nuclear system protection system (NSPS) were shown (see Figure 7). The changes are intended to improve reliability. The sensors of various parameters have been changed from essentially on-off devices to analog transmitters, protection functions can be easily and continuously checked by comparing meter readings with similar control parameters. The protection system can be tested on line and will utilize two out of four solid state logic, that is, two like variables must trip to cause the reactor to scram, the logic can be reduced to two out of three to permit maintenance on the fourth channel. The protection system is completely separate from the normal plant control system.

There will be four independent power supplies, one for each of the separate divisions. Mr. Grim described how the proposed system would operate with regard to inputs, outputs, and separation. The outputs of various sensors for a given parameter can be visually compared at the appropriate instrument cabinet.

GE has initiated a development effort that will extend over the next 2 or 3 years. This program will include reliability and equipment qualification.

The second major modification in the instrumentation and control system is in the rod control and information system. The primary function of this system is to effect control rod motion and the secondary is to obtain and display rod positions (see Figure 8). Part of the rod motion function is to restrict or prohibit certain rod actions. This system will replace the rod worth minimizer, the rod block monitor and the rod sequence control system. The new system will be designed to perform correctly or rod movement will be inhibited even in the presence of a single component failure. The system will be fully testable to verify correct operation at all times.

Each control rod is provided with two separate channels of rod position detectors, the information from these detectors is fed through individual

ASSUMPTIONS USED TO CALCULATE OFF-SITE DOSE CALCULATIONS AS A RESULT OF GASEOUS RADWASTE SYSTEM FAILURE.

| AEC EVALUATION ASSUMPTIONS* | DESIGN LIMITS (GE EVAL. ASSUMP) | EXPECTED |
|---|---|---|
| NOBLE GASES AT 30 MIN. DECAY, 350,000 μCi/SEC | 100,000 μCi/SEC | 25,000 μCi/SEC |
| 100% RELEASE | 10% NOBLE GASES<br>1% PARTICULATES<br>1% IODINES | 1.2%<br>1%<br>1% |
| ACCIDENT METEOROLOGY REGULATORY GUIDE 1.3 | VERY STABLE (1 METER/SEC WIND VELOCITY) | NEUTRAL (3 METERS/SEC) |
| SEMI-INFINITE CLOUD DOSE MODEL | FINITE CLOUD DOSE MODEL. | SECTOR AVERAGE FINITE DOSE MODEL |

*USE OF ANY THREE OF THE AEC EVALUATION ASSUMPTIONS WITH ONE OF THE DESIGN LIMIT ASSUMPTIONS RESULTS IN CALCULATED DOSES LESS THAN 10 CFR 20 LIMITS.

FIGURE 5

## DOSES RESULTING FROM FAILURE OF OFFGAS SYSTEM
(mRem/event)

| Dose to: | Guideline Dose | Site Boundary Distance - Meters A 300 | A 500 | B 300 | B 500 | C 300 | C 500 |
|---|---|---|---|---|---|---|---|
| Total Body-Gamma | 500 | 13 | 10 | 78 | 60 | 74 | 67 |
| Skin-Beta & Gamma | 3000 | 120 | 72 | 930 | 580 | 1200 | 490 |
| Lung | 1500 | $10^{-5}$ | $10^{-5}$ | $10^{-3}$ | $10^{-3}$ | $10^{-1}$ | $10^{-2}$ |
| Bone | 2800 | $10^{-2}$ | $10^{-2}$ | 1 | 1 | 14 | 8 |

Assumptions - 350,000 µCi/sec continuous release

A. 100% release from small vessels, 20% release from holdup pipe, 10% release from charcoal, very stable meteorology (1 meter/sec), finite cloud gamma model, no building dilution.

B. Same as A except 100% release.

C. 100% release, meteorology of Regulatory Guide 1.3, finite cloud gamma model, factor of 3 reduction for building wake.

*Figure 6*

## AFFECTED
## PROTECTION SYSTEMS

- REACTOR PROTECTION (SCRAM) SYSTEM
- NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM FOR ISOLATION VALVE CLOSURE (GE PROTION OF REACTOR VESSEL AND CONTAINMENT ISOLATION CONTROL SYSTEM)
- EMERGENCY CORE COOLING SYSTEMS
  - —AUTO DEPRESSURIZATION SYSTEM (ADS)
  - —HIGH PRESSURE CORE SPRAY (HPCS)
  - —LOW PRESSURE CORE SPRAY (LPCS)
  - —LOW PRESSURE COOLANT INJECTION (LPCI) MODE OF THE RESIDUAL HEAT REMOVAL SYSTEM (RHR)
- REACTOR CORE ISOLATION COOLING SYSTEMS (RCIC)

*FIGURE 7*

## ROD CONTROL & INFORMATION SYSTEM (RC&IS)

PRIMARY FUNCTIONS:

°Effect Normal Control Rod Motion (Not Scram)

Initiated by:

Operator

Restricted by:

Rod Positions (Rod Pattern)

Plant Status

°Obtain and Display Rod Positions

REPLACES AND/OR INCLUDES:

°Reactor Manual Control System

Rod Drive Control System (RDCS)

Rod Position Information System (RPIS)

Rod Worth Minimizer (RWM)

°Rod Block Monitor (RBM)

°Rod Sequence Control System (RSCS)

FIGURE 8

rod position information systems into two rod pattern control systems where a determination is made regarding the permissibility of rod movement, either individual or ganged rods. The determination would be based on a preprogrammed rod withdrawal pattern.

Conclusion (Open)

The Subcommittee did not caucus.

Mr. Stuart (GE) made a strong plea to have GESSAR on the November 1974 ACRS schedule.

Dr. Isbin explained that the Subcommittee was attempting to resolve as many issues as possible, including generic issues before bringing the project before the ACRS. He said that the Subcommittee would try to meet with GE again but he could not provide a date at this meeting. He advised Mr. Stuart that a date for the next meeting could not be established much before September 23, 1974.

Dr. Isbin pointed out a number of outstanding issues, including ATWS and compliance with Appendix K.

It was established that another Subcommittee meeting would be held before GESSAR would be brought before the full Committee.

* * * *

Documents Available to the General Electric Subcommittee
September 11, 1974

1. Agenda dated September 6, 1974.

2. General Electric Standard Safety Analysis Report and Amendments 1 through 18.

3. Table of AEC Regulatory Staff "Open Items on GESSAR" (Attachment A).

4. Viewgraphs used by General Electric Company.

5. AEC Regulatory Staff summary status of GESSAR open items.

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OPEN ITEMS ON GESSAR

1. Seismic and Quality Classification of Relief Valve Discharge Piping
   We say it should be seismic Category I, Quality Group C. GE agrees up to the first seismic restraint. From there on, they say non-seismic and Quality D. (RSB-3.2.1) minor

2. Offgas Quality and Seismic Classification
   We say seismic I and Quality D "augmented." GE does not agree. (RSB-3.2.1) major

3. Liquid Radwaste Quality Classification
   We say Quality D "augmented." (RSB-3.2.1) minor

4. Tornado Missile Velocities
   GE's analysis eliminates many missiles since they say the velocity is zero. We have assigned velocities to all missiles considered. (AAB-3.5) minor

5. Broken Pipe Motion
   We say broken pipe can "garden-hose." GE says it will move in a plane but have not adequately justified their position. (MEB-3.6) minor

6. Pipe Restraint Minimum Gap Size
   GE needs to document that the minimum gap is 6" or greater. (MEB-3.6) minor

7. Break Criteria for Piping Passing Through Containment
   GE needs to provide specific criteria for these pipes. Our criteria outlined in 1.xx for pipe breaks are acceptable if documented by GE. (MEB-3.6) minor

8. Drywell Proof and Leak Tests
   We require both. GE has appealed. We've rejected their appeal. (SEB-3.8.3) major

9. Analysis Methods for SSE and Steam Line Break for Internals
   GE needs to commit to provide more detail either in the FSAR or in a topical report covering this subject. For the PDA, all we need is their commitment. (MEB-3.9.1.5) minor

10. Upset Condition Loading Combination
    We will require GE to define the OBE plus upset transient loading combination for the upset condition unless they can justify that such a combination is not required. They think they can justify by time history method. (MEB-3.9.2) major

ATTACHMENT A

11. Seismic Qualification of Instrumentation and Electric Equipment
GE has previously tested to IEEE-344 (1971). We require that added criteria be used or GE commit to 1974 IEEE-344 (draft). (MEB-3.10) minor

12. Scram Reactivity Curve
GE needs to provide the "D" scram reactivity curve in the transient and accident analyses. They say they will have that completed by October 15, 1974. We need prior to PDA. (CPB/Physics-4.3.8) major

13. Appendix 4A Questions
GE needs to reply to our questions on the core power distribution study prior to the PDA. They say these responses will be in by September 27, 1974. (CPB/Physics-4.3.8) major

14. Physics Analytical Methods
We need further description of the methods used by GE prior to PDA (CPB/Physics-4.3.8) major We have no date from GE.

15. Verification of Physics Methods
GE needs to compare measured reactor characteristics to calculations and document uncertainties prior to PDA. No date has been proposed. (CPB/Physics-4.3.8) major

16. GETAB
GE has factored GETAB into GESSAR. We need to review the results. (RSB-4.4) major

17. S/R Valve Designs
We need a commitment by GE to provide cutaway drawings of the valves, justification of why they expect better performance, results of bench tests and commitments to surveillance program. This can be an R&D item if GE will commit to it. (RSB-5.2.2) minor

18. RHR Single Failure
Based on our interpretation of the GDC 34, the RHR system has to be able to bring the reactor to cold shutdown assuming a single failure. GE says they have <u>functional</u> redundancy. (RSB-5.5.7) major

19. RHR Switch to Spray Mode
The switch from ECCS mode to spray mode should be automatic. GE says manual switch is OK. (CSB-6.3.1) major

ATTACHMENT A

20. Suppression Pool Makeup System
This system is new and our evaluation is not complete. (CSB-6.2.1.4) major

21. External Drywell Design Pressure
We will require the drywell be designed for an external pressure of 21 psid since the drywell depressurization rate cannot be accurately determined. (CSB-6.2.1.5) minor

22. Containment Vacuum Breaker Sizing
GE has not presented complete information as to how they sized the containment vacuum breakers. We need added information on how heat sinks were handled and why only one spray train was considered operating. (CSB-6.2.1.5) major

23. Subcompartment Pressure Analysis
GE has not presented assumptions used in the analysis or results of calculations. (CSB-6.2.1.7) major

24. SGTS Separation
GE needs to verify that the components of the SGTS are redundant and separation is appropriate for an ESF. (CSB-6.2.3) minor

25. Pressure Analysis of Fuel Building, ECCS, and RWCU Rooms
GE needs to provide analyses that show that these rooms are maintained at a negative pressure following a LOCA (CSB-6.3.2) minor

26. Secondary Containment Bypass
GE needs to identify potential bypass leak paths and commit to periodically leak tests the paths. (CSB-6.3.2) major

27. Containment Air Cleanup
GE has not adequately justified his exceptions to positions C.3.d, C.3.1-e, C.3.h, and C.3.j of Regulatory Guide 1.52. (AAB-6.2.3.1) minor

28. Continuous Purge of Containment
GE will either have to provide internal filters and discontinue continuous purging or they will have to provide further design measures to assure rapid purge valve closure. (CSB-6.2.4) major

29. 8 x 8 Spray Distribution Test Results
GE needs to provide results of 8 x 8 spray tests to assure spray gets to all of the fuel. (RSB-6.3.1) minor

30. ECCS Reanalysis Assuming 2 LPCI Pumps Diverted to Spray
We need GE's reanalysis of the LOCA with two sprays diverted to assure ECCS performs acceptably. (RSB-6.3.1) minor

ATTACHMENT A

31. Post LOCA Manual Actions
We need a discussion by GE of what these actions are to assure that there are no undesirable consequences resulting from improper operator actions. (RSB-6.3.1) minor

32. Recirculation Valve Closure During a LOCA
If the recirculation valve closed during the LOCA the flow decay would be greater than pump coastdown as assumed in the present analysis. GE needs to show that valve closure is OK or that ESF grade equipment prevents such closure. (RSB-6.3.1) minor

33. ECCS per New Appendix K
This is in Amendment No. 19 and we need to review. (RSB-6.3.1) major

34. Pipe Insulation
GE needs to tell us the type of pipe insulation used in containment and demonstrate that it cannot foul the ESF strainers. (CSB-6.5) minor

35. RHR for SFP Cooling
We require that the plant be in the shutdown condition whenever the RHR system is used to assist in the cooling of the spent fuel pool. (APCSB-9.1.3) minor

36. RHR for SFP Water Makeup
GE needs another source of seismic Category I makeup. Use of the RHR, as this source, is unacceptable. (APCSB-9.1.3) major

37. SFP Cooling for Abnormal Condition
GE has only given the analysis of cooling for normal conditions. We need abnormal analysis. (APCSB-9.1.3) minor

38. Water Systems
Many of the auxiliary water systems do not have adequate descriptions, P&ID's or interface discussions to allow us to complete our review. (APCSB-9.2) major

39. MSLIV Leakage Control System
The review is not complete by us. We will need a discussion of how stem leakage is handled. (APCSB-9.3.2) minor

40. HVAC
There are several single failure concerns associated with auxiliary HVAC's. (APCSB-9.4.1.1) minor

41. Fuel Building Radiation Monitors
We feel these should be in the fuel building rather than exhaust ducting to detect potential activity puffs. (APCSB-9.4.6) minor

ATTACHMENT A

42. Gaseous Effluents
GE needs to provide additional means to reduce activity of releases since the SGTS is too small to handle exhausts from drywell and containment purge, sheild building, ECCS pump rooms and fuel building. (ETSB-11.3.2) major

43. Solid Waste Storage
GE only provides for a one month storage of solid wastes. We feel 6 months is more appropriate to reduce short-lived isotopes. (ETSB-11.4) minor

44. Free Water in Solid Wastes
GE needs to provide verification that there is no free water in solid wastes. Our concern is for leakage in shipment or storage. (ETSB-11.4) minor

45. PRT (PROMPT RELIEF TRIP)
GE has not told us why PRT is needed or what the alternatives are. (RSB-15.1) major

46. Further Staff Work
   a. EI&C Writeup - None exists to date for Chapter 7 and its impact on Chapter 15 (new designs)

   b. Other Accidents - We need dose calculations for refueling accident, etc.