STUDY OF QUALITY VERIFICATION AND BUDGET IMPACT

TASK FORCE REPORT TO THE DIRECTOR OF REGULATION

JANUARY 1974

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TASK FORCE REPORT

TO THE DIRECTOR OF REGULATION

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A. Purpose of the Study

A study of the reactor licensing process was completed by Regulation in December 1973 and a report entitled, TASK FORCE REPORT TO THE DIRECTOR OF REGULATION, STUDY OF THE REACTOR LICENSING PROCESS, DECEMBER 1973, was submitted for consideration of the proposals therein. The aforementioned study dealt with needs and methods of developing and implementing the designated site concept and any modifications to the Commission's standardization policy or to administrative procedures that might be desirable. The report of that study will be frequently referenced in this document and is hereby designated as reference A.

The purpose of this report is to discuss several possible variations of the recommendations contained in reference A, examine the operating and construction history of present day reactors and the Quality Verification programs of other agencies to identify potential improvements in the AEC's quality verification processes, identify potential long-term efficiencies in the licensing process, and delineate the budget impact of the recommendations set forth by this report and reference A.

B. Composition of the Task Force

The Task Force was chaired by L. V. Gossick, Assistant Director of Regulation; and vice-chaired by M. L. Ernst, Program Assistant to the Deputy Director for Reactor Projects. The Task Force members assigned to perform this study were: W. E. Campbell, Jr., Regulatory Standards; A. J. DiPalo, Office of Program Analysis; T. H. Essig, Technical Review - L; R. D. Smith, Fuels and Materials - L; J. H. Sniezek, Regulatory Operations; and S. A. Varga, Reactor Projects - L. Mr. M. G. Malsch, Office of the General Counsel, was assigned as part time legal advisor.

C. Definitions

The following definitions have been used within this report:

1. Balance of plant - the portion of the plant that is outside of the safety complex. Typically this includes the ultimate heat sink, cooling water supply systems, turbine-generator systems, and the switchyard.
2. Certification - a written statement attesting that a quality verification has been made.
3. Designated site - a site that has been through all the required administrative and public review processes, has been judged to meet all of the applicable safety and environmental siting requirements, and has been designated by the AEC as an acceptable site for a nuclear power plant having prescribed performance characteristics.

4. Design certification - a certification, following a design review, that all elements of the design important to safety have been considered; the design should provide an acceptable level of risk; and the specified reliability should be achieved, if appropriate design standards and specifications involving manufacture, fabrication, installation, construction and/or operation are met.

5. Level of risk - a probabilistic approach to predicting the effects of accident chains. The probability of occurrence of an event is multiplied by the defined consequences of that event.

6. Safety Complex - all the structures, systems, and components within the boundaries of and including containment, auxiliary building, control building, diesel generator building, and radwaste building, as defined by the AEC to be appropriate for PWR's and BWR's.

7. Product certification - a certification, following an inspection of appropriate portions of manufacturing, fabrication, installation, and/or construction, that appropriate design standards and specifications have been met.

8. Quality assurance - all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service.

9. QA inspection - the inspection by the AEC or its designee, of the QA program of a utility, NSSS, AE, constructor, subcontractor, or vendor in sufficient scope and depth to provide reasonable assurance that the QA program is consistent with the QA program plan; sufficient and adequate QA procedures have been written, approved, and distributed; and the QA program has been appropriately implemented in accordance with the QA procedures and the criteria of 10 CFR Part 50, Appendix B. This would include the sampling of work activities covered by the QA program. QA inspection includes the requirements of QA review.

10. QA review - the review by the AEC or its designee, of the QA organization and of the QA program plan of a utility, NSSS, AE, constructor, subcontractor, or vendor to assure that the QA organization and the documented scope and depth of the QA program is adequate to meet all of the criteria of 10 CFR Part 50, Appendix B.

11. Quality verification - any process by which defined attributes of a particular structure, system, or component are determined to have been met. This assurance may be obtained by direct or indirect inspection, review of design, qualification testing, operational testing, or any combination thereof.

12. Reliability - the characteristic of an item expressed by the probability that it will perform a required function under stated conditions for a stated period of time.

13. Safety-related - those structures, systems, or components that are subject to 10 CFR Part 50, Appendix B.

14. Standardization - the limitation of design, construction, fabrication, and/or installation options in regard to the safety related structures, systems, and components of light water nuclear power plants. As used in the "Recommendations" section of this report, this would include plant layout, line routings (piping and instrumentation), equipment location, and detailed functional performance criteria of individual components and structures comprising the safety complex.

D. Scope of Study

To accomplish the objectives identified in section A of this report, the work of the Task Force was divided into two major phases. Phase 1 consisted of the accumulation of factual information without necessarily providing specific recommendation or judgements. Phase 2 consisted of a detailed discussion of: all the information obtained during Phase 1, recommendations of the individual Task Force members and of members of the Regulatory staff, the recommendations contained

in reference A and in this report, and budget and legislative impact of these recommendations. Phases 1 and 2 were separated to gain as much objectivity as possible during the development of the factual information. Section F of the report discusses level of risk of nuclear power plants in a qualitative way and how confidence levels may be increased and Section G provides the Task Force's conclusions and recommendations in the various areas reviewed by the Task Force.

The deliberations of the Task Force were extensive, and there were numerous pros and cons identified for the various options considered. Condensations of some of these deliberations are provided in enclosures to this Task Force report. They provide some of the rationale for the recommendations made by the Task Force, both in this report and in reference A, and include some of the variations that can be made to the basic recommendations and their impact as to effectiveness, reasonableness, and budget. These enclosures are as follows:

Enclosure 1 - discusses the subject of verification of quality, techniques that may be employed to verify quality, and the concept of certification.

Enclosure 2 - describes various selected incidents and problems that have occurred at nuclear power reactors during the past several years.

Enclosure 3 - provides the budget impact of the Task Force's recommendations through fiscal year 1985, as compared to the baseline budget assuming "business as usual" through fiscal year 1985 and as compared to an expanded baseline which considers increasing the levels of review and inspection as recommended by the Task Force without obtaining the efficiencies that would be accrued from the prompt implementation of site designation and plant standardization.

E. Summary and Conclusions

This report and enclosures thereto contain a number of recommendations for the development and implementation of changes to improve the quality verification processes both on the part of the nuclear industry and the AEC. Upon full implementation of the recommendations contained herein, two objectives should be achieved. First, plant reliability should be increased; and second, increased assurance should be provided that the aggregate level of risk will not increase as nuclear power plants proliferate in coming years.

The results of the current AEC inspection program, the number of operating problems associated with nuclear power plants, and the need to assure that the aggregate level of risk is maintained at an

acceptable level as the number of nuclear power plants increases, argue for an increased quality verification effort on the part of both industry and the AEC. Consequently, the Task Force believes that the Regulatory staff's inspection programs should be expanded. It is felt that augmented qualification testing of components should be required of industry; independent testing of selected components should be performed by AEC-certified laboratories; direct inspection of NSSS vendors, AE's and component vendors should be conducted by the AEC; and standardized plant design configuration control, plant layout, line routings, and component selection should be examined by the AEC during the construction phase. Likewise, industry should be required to demonstrate an improvement in quality assurance pertaining to design, construction, and operation. These steps should not only assure greater plant safety and reliability, which will be mandated when 20-50 percent of the electric energy market is nuclear, but it should also enhance public acceptance of nuclear power.

Additionally, this report provides budgetary information that is fully supportive of the recommendations set forth in this report as well as those in reference A regarding standardization and shortening of the licensing and construction processes.

F. Level of Risk

1. General

A discussion of level of risk and its impact on the Regulatory process is one of the primary themes of this Task Force report. The following discussion portrays the rationale for many of the Task Force's recommendations regarding scope of review and quality verification.

In a technologically oriented society, many activities have an associated risk that must be balanced against its benefit to the public. As a practical matter, a new technology introduces new risk, although it may also decrease other existing risks. This new risk is more readily quantified as more information is obtained and is often reduced as technology improves. A difficulty arises, in some cases, where freedom from accidents is not an adequate demonstration, statistically, of a sufficiently low level of risk. This is especially true in an emerging technology where there is a requirement that accident probabilities be very low and a broad base of satisfactory operating history has not yet been established.

Regulation of nuclear power has been based on conservative practices and, thus far, appears to have been successful. In over 180 commercial reactor-years of operating experience,

there exists a perfect record of public safety; that is, not a single accident has interfered with activities of people in public areas. In addition, military power reactors have demonstrated over 1500 reactor-years of safe operation. Regulatory policies have continued to evolve and have stressed the importance of assuring safe operations, but methods for estimating the quantified degree of safety (or the level of risk) of a nuclear power plant and of the nuclear power industry are still under development. This does not mean that the level of risk is completely unknown. Much qualitative and quantitative information is available about postulated accidents in nuclear power plants. Applicants' safety analysis reports and Regulation's safety evaluations contain quantitative discussions of postulated accidents. Recent and ongoing Commission studies are considering the public risk due to both radioactivity released in effluents and postulated severe accidents, as well as comparing the risks and benefits of various technologies for generating electrical power.

The determination of an acceptable level of public risk is actually a matter which should be debated and established in the public arena. It is a social, economic, and political question which cannot be resolved solely by a regulatory or technical decision. It is recognized that some of the technical

issues are difficult for the layman to absorb quantitatively, especially as related to the occurrence rate of low probability events. In such cases involving very low probabilities, the level of risk is difficult for even the engineer to quantify.

As a new technology develops, it will intuitively (and rightly so) be compared by the public with known existing risks (e.g., automobile, drowning, fire, etc.). To assist in the comparative process, these risks can be presented in numerical terms. For example, based on an examination of all categories of accidental death statistics for 1966, accident types resulting in a probability of death greater than 10^{-4} per person (one person in ten thousand) per year are difficult to find (automobile deaths are the notable exception, with an annual rate of about 2.7×10^{-4}, or about 2.7 deaths per ten thousand people per year). This level of risk is unacceptable to many people; witness the public demand for action to reduce the automobile risk. By contrast, accident types with risk levels of 10^{-6} (one in a million per year) and lower are normally of little concern to the public.

The public assessment of risk is a value judgment which appears to be influenced by, among other things, the number of people killed in a particular accident and the amount of accompanying property damage. For example, a plane crash killing 100 people

with an associated property damage of 100 million dollars would receive more publicity and thereby create greater public concern than accidents whereby people die singly, such as by mechnical suffocation (approximately 1300 deaths per year).

The risk to the public arising from a given nuclear reactor, or from all nuclear reactors, is the sum of the risks from all operating modes, including low-probability postulated accidents. (The risks, for example, from mining the fuel, transporting the fuel and the radioactive wastes, and storing the wastes are not considered in this paper, but are the subject of other evaluations, including ongoing rulemaking proceedings.) A simple way of putting a number on this risk is to express it as the total of all risks which result in a degradation of the human environment from all conceivable operations and accidents, each weighted by its respective probability of occurrence. However, quantification of these risks is incomplete, if identification of all accident combinations with significant nonzero probabilities is not accomplished.

The accidents of interest are divided into two categories: 1) those due to random, independent failures in reactor systems (a single failure is protected against); and 2) concurrent related

failures of components or systems due to some common design, construction, or operation error or to a common influence such as an earthquake or tornado. The accident involving random independent failures may be evaluated by the use of accident probability chains. These chains are used to break down accident sequences of low probability into higher probability constituent events. The higher probability events can be studied based on experience and engineering. The probability of occurrence of the accident chain, too low to estimate based directly on experience, can then be calculated from the probabilities of the constituent events and the ways in which they can combine to constitute the postulated accident.

This method is also applicable to nonrandom or common-mode failures, where these failures are known or can be estimated; although risks calculated in this way usually do not include components attributable to nonrandom failures. It is difficult to assure completeness in regard to common mode failures.

While accident chains can be postulated and the appropriate probabilistic equations written, the availability of actual performance information (in the form of reliability data) is a matter which has not yet been well addressed by the AEC or the nuclear industry. In some cases data are available, in

others data are available only in a qualitative form, and in still others experimental data of some relevance are available. Where data are missing or obscure, the needed predicted probabilities must be derived through an appraisal of the available data and values suggested by persons knowledgeable and experienced in these areas. Therefore, efforts should be made to improve fault information systems so as to collect reliability data which will make these analyses more precise.

Many general and specific fault information systems are available. Systems such as the Government-Industry Data Exchange Program (GIDEP) are designed to collect data on small, generally available (off-the-shelf) components; whereas systems such as those employed by RRD and the AEC Nuclear Safety Information Center are designed to collect reliability data related to a specific area of nuclear activity. The Edison Electric Institute is presently instituting the Nuclear Plant Reliability Data (NPRD) System. The NPRD System is specifically designed to collect failure rate data on certain specific safety-related nuclear power plant components and equipment.

In addition to utilization in the quantification of level of risk, fault information systems may be effectively used to ascertain the areas meriting increased emphases in regulatory

review and standards development; identify areas where increased qualification, preoperational and functional testing is required; and provide a logical basis for assignment of inspection manpower.

2. Current Level of Risk - Safety

Review of the operating history associated with 30 operating nuclear reactors has shown that during the period 1/1/72 - 5/30/73 no nuclear accidents occurred and no member of the public was injured in any way due to radiological causes. However, this record also contains approximately 850 abnormal occurrence reports filed with the AEC. While the vast majority of these abnormal occurrences represented failures that are anticipated, will always occur with manufactured equipment, and are protected against by the redundant design of nuclear systems; and while none of them resulted in a significant direct threat to the health and safety of the public; many of the occurrences either illustrated failures in QA programs during the construction or fabrication phases or were symptomatic of or identified design weaknesses in safety-related components and systems. Many of the occurrences also were of a generic nature requiring followup investigations at other plants.

Forty percent of the reported occurrences were traceable to some extent to design and/or fabrication related deficiencies. The remaining occurrences were caused by operator error, improper maintenance, inadequate erection control, administrative deficiencies, random failure, and combinations thereof. Examples of occurrences that exhibited significant safety importance are provided in Enclosure 2.

In view of the increasing number of operating reactors which will be on-line in the 1980's and 1990's; and considering the large number of reportable safety-related occurrences, coupled with the fact that many of them were generic in nature and were not appropriately identified during the actual design, qualification testing, fabrication, erection, and preoperational testing phases of these reactors; one could conclude that current AEC review and inspection practices need improving for the future, primarily in areas where little or no review or inspection is currently performed.

The present state of the art does not permit exact quantification of the present level of risk from nuclear reactors nor does it provide for the quantification of improvement in the level of risk which might arise from proposed alternative modes of operation within Regulation. The Commission safety study now underway

should provide substantial additional quantification and a developed technology for obtaining such quantitative information. It will, however, be beneficial to continue to use operating experience to improve these reliability numbers and to continue the search for significant postulated accidents that may have been overlooked.

The Regulatory staff has recently suggested ("Technical Report on Anticipated Transients Without Scram" WASH 1270) a goal for risk level of nuclear reactors; that the occurrence rate of accidents with consequences worse than 10 CFR 100 in the entire USA be held to less than once per thousand years. If the typical such accident were to kill somewhere between 100 and 1000 people (in the range of a bad aircraft accident), then an average risk to a US citizen of dying from such accidents would be about one in a billion per year - a negligible risk when compared to other risks discussed above.

Today, with 40 reactors operating, this goal requires each reactor to have a probability of less than one in 40,000 per year for experiencing a bad accident. Within the next 30 years, there will be on the order of 1000 operating reactors, and the goal for each reactor in the future should therefore be one in a million per year for such accidents.

The Task Force believes that the Regulatory practices are adequate to provide reasonable assurance as to the safety of present-day nuclear reactors and to protect the health and safety of the public, considering the numbers of reactors. The bases for this conclusion are the redundant designs utilized, the defense-in-depth concept, and the continuing improvements that have been made in the design of reactors and the assurance of quality. However, considering the large number of reactors predicted to be operational by the year 2000, and in view of the overall incident record over the past several years combined with the common mode failures that have been identified, the Task Force believes that further continuing actions need be taken to provide additional assurance that the probability for such an accident will be one in a million or less per reactor-year.

The Task Force thus concludes that Regulation should strive to implement the various programs identified elsewhere in this report that are designed to increase the confidence level of reactor safety and enhance public acceptability. For the most part, these programs are aimed at areas where little or no effort has been expended in the past. There is very little in this recommendation that represents an increase in the depth of review or inspection.

3. Current Level of Risk - Environment

At the present time the Regulatory staff believes that the amount of AEC effort in NEPA reviews is appropriate to provide reasonable assurance that the environment is being duly protected. The AEC appears to be living up to the requirements of NEPA and, while there are minor weaknesses in the AEC environmental review process, these have been noted and will be appropriately addressed in future NEPA statements. For example, two weaknesses currently most often referred to are: nonnuclear accidents and socio-economic impacts on the local economy resulting from reactor plant construction forces.

4. Current Level of Risk - Safeguards

At the present time, requirements for control and accounting of special nuclear materials are contained in 10 CFR Part 73, "Physical Protection of Special Nuclear Material," and 10 CFR Part 70, "Special Nuclear Material." To assist reactor licensees in the preparation of their security plans to protect against sabotage from malcontent employees, dissident groups, etc., the AEC has issued Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage," 1973. In the opinion of the Task Force, any foreseeable proposed safeguards program for reactors would not impact significantly on

regulatory staffing nor would it have any significant effect on the safety design of the reactor.

5. Ways to Improve the Level of Risk

It was concluded by the Task Force that it is possible to identify areas now where changes in the regulatory process may be made which should, in principle, decrease the level of risk or improve confidence in the level of risk value. Improvement in these areas is not necessarily capable of being measured and translated today into numerical indicators of change in risk; but judgements can be made as to whether or not significant improvements could be made. Areas the Task Force believes are amenable to improving the level of risk are listed below and were considered by the Task Force to warrant increased regulatory attention. Detailed recommendations in these areas are provided in Section G.

a. Quality Assurance

Quality assurance provides a means of control and verification whereby those responsible for project management and for Regulatory review and inspection can increase their assurance that the quality required for safe, reliable, and efficient operation will be achieved. QA covers all aspects of the project and includes, fo example:

(1) management and planning

(2) design and development

(3) procurement

(4) manufacturing, fabrication, and assembly

(5) construction and installation

(6) operation and maintenance.

The Task Force concludes that based on the numbers and types of incidents and problems that have occurred over the past two years, the industry QA programs and the AEC QA inspection programs should continue to be upgraded for operating reactors, reactors under construction, NSSS and A-E engineering offices, and vendors.

b. Depth and Focus of Application Review

The depth of AEC review required to provide adequate assurance of quality is not easily determined. The AEC review philosophy to date has been to probe rather deeply into areas of safety importance, but to avoid designing any portion of the plant for the applicant. The current AEC review effort is about 2% of the engineering effort required to design a plant. The Task Force concludes that the AEC review philosophy is good, and this effort should be appropriately expanded in the

areas of plant layout, equipment location, line routings, systems performance and interaction, and establishment of detailed performance specifications. Also, designs should be standardized as much as possible to enhance safety.

In addition, the techniques and procedures developed by the Rasmussen study should be carefully studied from the standpoint of their applicability to comparative evaluations during the review process. It is conceivable that an improvement and further standardization in our safety evaluations utilizing these techniques and procedures could be effected, and a quantitative assessment could in this way be made of the significance of changes or "improvements" that are proposed. Not only would this aid in our safety evaluations, but it would continually update the experience and statistical data being accumulated.

c. Qualification, Acceptance, and Operational Testing

The Task Force concludes that the level of confidence that structures, systems, and components will perform as required is a strong function of the adequacy of qualification, acceptance, and operational testing programs. The designs should be modified as dictated by the results of test data and requalified, as appropriate, to assure

performance in accordance with design requirements. The Task Force recommends that increased effort to develop standards be applied to this area; a survey be made of industry and government testing facilities to determine their capabilities to test components and systems under varying operating conditions and environments; more qualification testing be conducted by the industry, as appropriate, particularly for components critical to reactor safety; and selected components be qualification tested (certified) in AEC-approved laboratories.

d. Inspection

To date the Regulatory inspection philosophy has been focused primarily on obtaining assurance that the applicant or licensee (utility) is implementing an adequate quality assurance program during the construction and operation of his plant. The inspection effort by Regulation is performed on a sampling basis and encompasses only about 1-2% of the safety related activities that take place on the construction site. The Task Force concludes that the AEC inspection philosophy is good, and this effort should be expanded significantly in the area of engineering, fabrication, and

construction to include areas not currently being inspected. Inspection of preoperational testing activities should also be expanded.

e. Enforcement

The Task Force concludes that any regulatory policy regarding the safety and integrity of the final product will be strongly enhanced by a strong forcing function, namely enforcement. A strong enforcement program will tend to reduce total plant erection schedules (in the long run) and decrease the level of risk, since there would be less inclination by utilities to make unnecessary design changes or to accept equipment that has not been rigorously tested. The Task Force concludes that, in cases of unacceptably poor performance of safety related activities, the activities should be promptly halted via appropriate enforcement action.

f. Standards and Criteria

The Task Force identified a need for expedited standards work in the areas of qualification, preoperational, and operational testing, piping layout, siting criteria, plant performance criteria, and quality assurance.

g. Assessment of Risk

The Task Force concludes that the collection and evaluation of reliability (design, test, operational) data is an essential and integral part of the U.S. nuclear reactor program and should be augmented within Regulation. The Task Force concluded that studies to evaluate the level of risk are important, would have materially helped the Task Force in its deliberations, and should be initiated within Regulation.

G. Recommendations

1. General

The recommendations of the Task Force regarding the various matters studied are contained in this section of the report. The rationale behind the recommendations may be found in the enclosures to this report, and in some instances, in reference A.

While most of the recommendations in the site designation and standardization areas, as set forth in reference A, are near term considerations that deal with speeding up the licensing review process and making it more efficient, there are some manpower increases in the licensing area that are required to increase the scope of review of design in several areas not

previously addressed in sufficient detail. These increases are needed to assure a continued acceptable level of safety and to implement standardization.

In the inspection area substantial increases in inspection manpower are recommended by the Task Force; not to shorten the process or to increase long term efficiency, but to provide additional assurance regarding the level of risk. The Task Force believes that implementation of these recommendations should be considered very carefully, since they represent a very nominal auditing effort when considering the total manpower expended by the nuclear industry in the design, fabrication, construction, and operation of nuclear power reactors.

The additional AEC effort should result in a significant increase in our confidence level as to the safety of nuclear reactors. It must be emphasized that, in general, depth of review is not being increased; rather, areas not previously examined would be inspected. Examples of such areas are line routings, compartmentalization, configuration control, foundations, containment penetrations, and qualification testing.

An additional point that must be considered is that, in the future, when nuclear power will represent 20-50% of the total electrical generating capacity in the United States, there

would be a severe economic and social penalty, if nuclear power reactors were shut down or derated because of generic problems, such as the ones that have occurred recently. While problems normally can be identified and rectified before they become serious and immediate threats to public health and safety, sometimes shutdowns or deratings may be required to maintain a sufficient margin of safety while necessary corrective actions are being taken. While such power outages can be assimilated relatively easily today, when nuclear power represents less than 5% of the total generating capacity, increased reliability of plants and freedom from generic problems that could impact regional power production must be achieved by future generation nuclear power plants. A significant catalyst for these improvements is improved QA and Regulatory attention through inspection.

2. Quality Verification

The following are specific recommendations made by the Task Force in the area of quality verification.

It is recommended by the Task Force that AEC certification not be considered by Regulation at the present time, except for the "certification" of standardized designs. It is concluded that a sufficient case has not been made at present to support the

concept of vendor certification or of a comprehensive program of component certification. Even if a comprehensive component certification program had been deemed to be necessary, standards have not been developed sufficiently in the areas of concern to be able to define clearly the requirements that must be met by the certified products. The Task Force believes that this question should be reviewed again at a later date; and if a comprehensive program of vendor or component certification is deemed to be warranted and if our requirements for certification are clearly spelled out, then we should proceed on such a course.

Regarding the certification of certain components, the Task Force does recommend that AEC approved laboratories be established to perform mandatory qualification tests on certain key components and to perform qualification tests on a sampling basis on other components of safety importance. The sampling programs and necessary standards should be developed by the AEC, and the costs of the tests should be paid for by the manufacturer. Purchase specifications or contracts, as appropriate, should require the performance of these tests. The AEC should inspect these laboratories and should be able to withdraw approval of their program for good cause.

Although the Task Force does not recommend the establishment of a comprehensive component certification program at this time, the Task Force did define several areas that should be improved in order to have a reasonable level of confidence that the quality of a particular activity has truly been verified. These areas involve a) improvement in qualification and operational testing, b) more efficient and effective action on generic problems, c) the establishment of an effective QA inspection program for NSSS's, AE's, and vendors, d) more direct access to NSSS's, AE's, and vendors for inspection purposes, and e) increased and broadened inspection at construction sites, primarily in areas not currently being inspected.

Regarding the inspection of vendors, it is concluded that Regulation currently does not really have a vendor inspection program, since only a few man-years were expended in this area in FY 73. It is concluded that this should be increased considerably (to approximately 72 man-years each year). The technique followed should be similar to the one used today; i.e., QA inspections with increased attention paid to problem areas. The AEC should require that licensee, NSSS, and AE contracts or purchase orders include a boilerplate requirement which would authorize unannounced AEC entry into vendor shops for inspection

purposes. Generic problem solving on "standardized" items should be pursued by the standardized design "certificate holder" and the AEC should take action directly with these "certificate holders". Enforcement action on "non-standardized" items should be taken through the utility applicant, as is done today. As an alternative, legislation permitting direct access into vendor shops and enforcement of AEC requirements at the vendor level could be put forth.

Components should be appropriately qualification tested. Improvements should be made in the qualification testing specifications listed in purchase orders, and toward this end our standards efforts in determining qualification testing specifications should be amplified. Also it is concluded that the licensee or "certificate holder" should require in their contracts or purchase orders that subcontractors or vendors certify to the purchaser that the identified qualification tests have been satisfactorily performed and should provide a description of the test conditions and results, as appropriate.

Regarding the AE's and NSSS's, the AEC has no current ongoing inspection program. It is recommended that the AEC perform routine QA inspections of these organizations. Additionally, engineering

office inspections should be made to assure that standardized projects are really being handled on a standardized basis. Inspection of the AE's and NSSS's should require an expenditure of approximately 18 man-years of professional effort each year.

In the area of construction and preoperational testing, it is concluded that the AEC inspection base for establishing our level of confidence must be increased. The construction inspection force should be more than doubled; and improvements in efficiency and coverage can be obtained if resident AEC inspectors are utilized. This increase in effort would permit inspection in areas that had to be deleted due to lack of manpower in the past several years, such as soils, site preparation, foundations, containment penetrations, and periodic work performance reinspections in areas originally found to be adequate. It would also provide for increased effort in the inspection of plant layout, equipment location, and pipe and line routings to support the concept of standardization. The inspection of preoperational testing, operating procedures, emergency plans, and health physics aspects should be increased by about 50% over the present effort to provide broader coverage in these very important areas.

Regarding operating reactors it is concluded that a nominal increase (approximately 20 professionals) in the manpower expended in reactor operations is warranted, and that in some instances resident or area inspectors could be utilized to increase efficiency. The nominal increase in inspection effort that is recommended would be in the areas of management QA inspection programs, a more detailed examination of training programs for operators and maintenance personnel, and the development of more uniform and effective operating procedures. It should be recognized that the subject of operating reactors was not pursued in depth by the Task Force, and while general discussions held with knowledgeable persons in Regulation, combined with the opinions of the appropriate Task Force members, did not reveal any major problems with the level of inspection in the remaining portions of the operating reactors inspection program, an audit of this program is currently being conducted by RO and could well affect the above consensus.

The AEC should make use of third party inspections wherever appropriate, for example, in the insurance inspection of the pressure integrity of ASME Section III vessels, pipe, pumps, and valves. However, if such third party inspections are relied upon by the AEC, assurance must be obtained that the

inspection agency is qualified, is inspecting to AEC standards, has sufficient independence, has sufficient documentation of its program and of its inspection results, and will permit AEC audit of its activities.

3. Assessment of Risk

It is recommended that the fault information systems needed by Regulation be emphasized programmatically, and the AEC should be prepared to use the information available from the NPRD system and other appropriate data collection systems to more effectively carry out its regulatory functions. These systems are required if we are to do a better job in the future of evaluating the level of risk of nuclear reactors, and these evaluations must be effectively fed back into the regulatory process.

Extension of the Rasmussen-type study by Regulation, or direct access to such continuing studies as may be conducted by the General Manager, should be considered for other accident modes and other reactors. While this type of analysis is not a panacea with respect to evaluation of the safety of reactors, these techniques will help to identify weak areas in design, fabrication, construction, and operation and will give a reasonably good numerical value of level of risk.

4. Legislation

Legislation may be required to facilitate AEC inspector acess to the shops of non-licensed suppliers of components, architect-engineers, and NSSS vendors. Likewise, to facilitate enforcement of required quality verification at the most effective level (namely the product supplier), legislation should be recommended.

5. Improvement and efficiency

As part of the foregoing recommendations, the task force has not only identified areas where more work needs to be done but has also identified areas where future efficiencies can be achieved. These efficiencies are briefly discussed below.

a. Regarding the inspection of construction sites, NSSS's, AE's, and vendors, the required Regulatory Operations inspection would be considerably more comprehensive than that which now exists. However, the utilization of resident inspectors at construction sites could result in a considerable improvement in efficiency. As standardization comes into full usage, there will undoubtedly be some inspection efficiencies that can be accrued, plus the construction time will be reduced thus requiring somewhat less inspection. Also, many of the sites will have two or more facilities being constructed

at one time, thus permitting further efficiencies. To estimate the efficiencies that could be achieved at multiple construction sites, the Task Force assumed that only half of the construction inspection force would be required for subsequent units under construction at the same site, as compared to single site construction needs. Also, it was assumed that there would be a 5% savings per year for 5 years in the inspection of vendors, NSSS's, and AE's, starting in FY 1980, to reflect improvements in QA programs for those organizations.

b. Regarding operating reactors, the Task Force did not study this area in great detail, however it does have recommendations as to achievements of efficiencies in the future. It appears that a possible management goal could be the improvement of efficiency by approximately 4% a year for a period of 9 years, starting in 1977. This would apply to RO, RP, and TR effort. This would mean that by 1985 the manpower expenditure per reactor should be approximately 64% of what it is today. While this type of improvement and efficiency was assumed by the Task Force in the determination of budget impacts through 1985, it was felt that a study of operating reactors should be performed in detail to

determine whether or not these recommended improvements and efficiencies are really attainable. The judgements that lead the Task Force to the conclusion that such efficiencies could be achieved are: 1) there should be a better definition of inspection requirements over the next few years which could be factored into the existing program; 2) the impact of standardized plants, starting in the early 1980s, should allow a reduction in the manpower expenditure per plant due to similarities in equipment, plant layout, and operating procedures; 3) the impact of standardization and the expanded Licensing review in specified areas should result in a reduction in the number of operating problems that are encountered in individual plants; 4) the increase in number of operating power plants should increase the density of these power plants in the country and should result in the establishment of sites containing two or more plants that are relatively close together, which would permit the utilization of resident or area inspectors to increase the efficiency and effectiveness of these inspectors; 5) the emphasis on utility QA over the next few years should make a marked improvement in the operation of nuclear power plants; 6) the number of licensees will not increase as fast as the number of plants,

since more and more licensees will be owning multiple plants; and 7) if emphasis is placed on operator and maintenance personnel training and also on plant standardization, it is concluded that operating errors should be decreased somewhat thus allowing a reorientation of inspection priorities in this area.

H. Budget Impact of Recommendations

The Task Force has made a detailed analysis of the budgetary impacts of the recommendations set forth by this report and reference A. The analysis summary is presented in Enclosure 3 of this report.

To assist in the benefit-cost analysis of the Task Force recommendations, a detailed baseline budget was constructed against which the impact of the recommended changes could be compared. The baseline budget for Technical Review, Reactor Projects, Regulatory Standards, and Regulatory Operations was created using the basic work units established by these groups and applying them to the projected work loads through 1985. The result is a linear extrapolation of the expected expenditure of professional manpower, primarily due to the basic licensing and inspection tasks involved regarding CP's and OL's. It does not account for the overhead associated

with the organizations, such as management, supervision, administrative, or clerical efforts. It also does not account for backlog and for work which is peripheral to the basic activities covered, such as assistance to other organizations, solving generic problems, and other non-case related effort. Thus, before the presented budget data may be compared to FY 74 or '75 budget submittals or converted into total manpower requirements, appropriate adjustments must be made. It was not within the scope of this report to make such adjustments.

The basic activities of the Directorates were examined individually in light of the Task Force's recommendations. For each, data were presented which show the projected baseline budget for the activity and the budget impact resulting from recommendations of the Task Force regarding the activity. The resulting incremental changes are considered to be the most important information provided in this section. The cumulative incremental changes for the organizations are summarized to show the net effect on Regulation of implementing the recommendations.

The recommendations of the Task Force, as expressed in Section G of this report and in reference A, are designed to streamline the licensing process, reduce the time required to license and construct

nuclear power plants, and increase reliability and the confidence level regarding reactor safety. Due to the history and types of problems associated with nuclear plants (Enclosure 2), the Task Force concludes that the scope of participation by the AEC in the areas of design review and quality verification should be increased, even if the recommendations set forth by the Task Force in reference A are not adopted. It should be pointed out that essentially the same confidence level regarding reactor safety should exist whether or not the Task Force's recommendations regarding site designation and plant standardization are adopted. However, as discussed herein, the Task Force recommendations also provide many long term budgetary efficiencies due to standardization and site designations, as well as efficiencies in the licensing process.

As pointed out above, the baseline data represent doing business as usual - no site designation or standardization and no augmentation of review or inspection effort or of standards development. The Expanded Baseline, however, presents the manpower augmentation that the Task Force feels would be required to implement a satisfactory scope of technical review and inspection, considering the large numbers of reactors predicted to be in operation by the year 2000. The Expanded Baseline includes such things as expanded review effort on line routings

and plant layout; increased construction inspection program; implementation of a NSSS, AE, and vendor inspection program; an increased operating reactor inspection program; increased emphasis on reliability and level of risk evaluations; and needed standards development to support the above areas. The Recommended manpower curves consider all the above, but they also include the near-term manpower required to implement the site designation and plant standardization concepts and the long-term efficiencies to be accrued from such concepts.

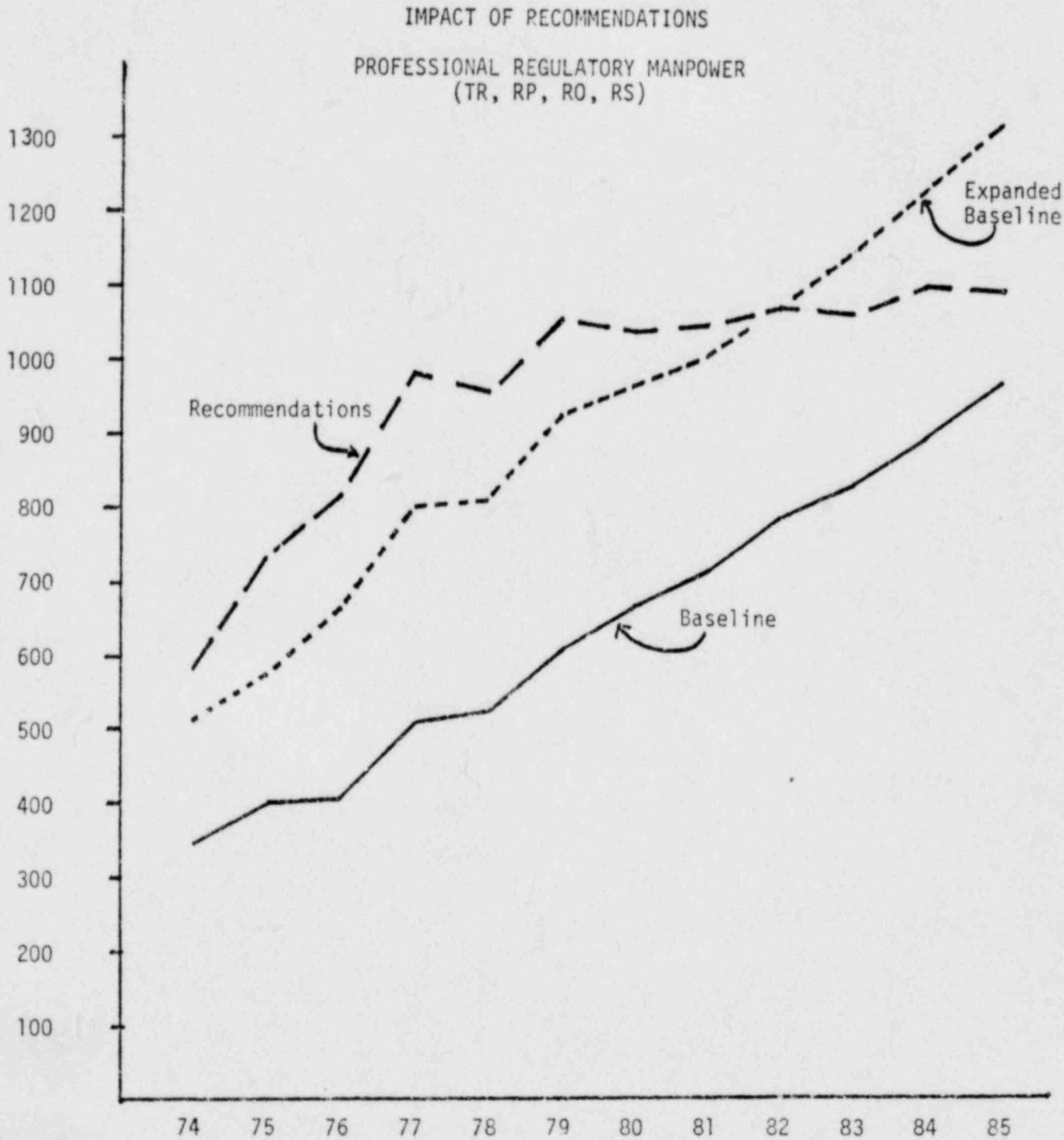
The budget data and efficiencies developed in Enclosure 3 are summarized and presented graphically in the following figures.

Figure 1 depicts the overall effect on Regulation in terms of applied professional end of year staffing levels for the period 1974-1985. The base-line, expanded baseline and recommended curves represent the composite of corresponding curves shown in Figure 2 for Technical Review, Figure 3 for Reactor Projects, Figure 4 for Regulatory Operations, and Figure 5 for Regulatory Standards.

Upon examination of Figure 1 several facts are quite evident:

1. The Task Force recommendations, when compared to the Baseline projection, will require a significant increase in resources,

Figure 1


IMPACT OF RECOMMENDATIONS
PROFESSIONAL REGULATORY MANPOWER
(TR, RP, RO, RS)
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
Expanded Baseline
Recommendations
Baseline
74
75
76
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84
85

Figure 2

IMPACT OF RECOMMENDATIONS

TECHNICAL REVIEW

TECHNICAL MANPOWER

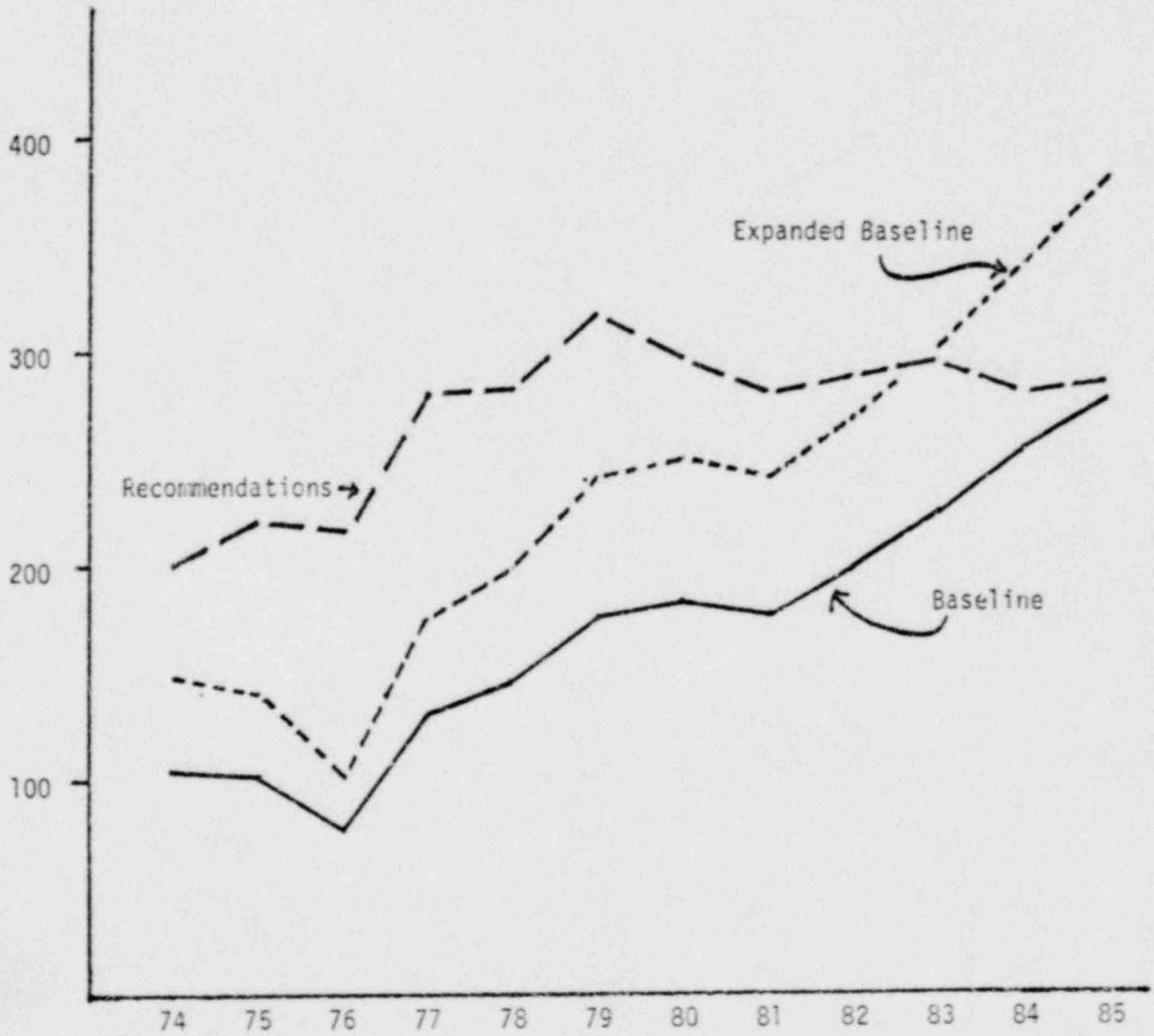


Figure 3

IMPACT OF RECOMMENDATIONS

REACTOR PROJECTS
TECHNICAL MANPOWER

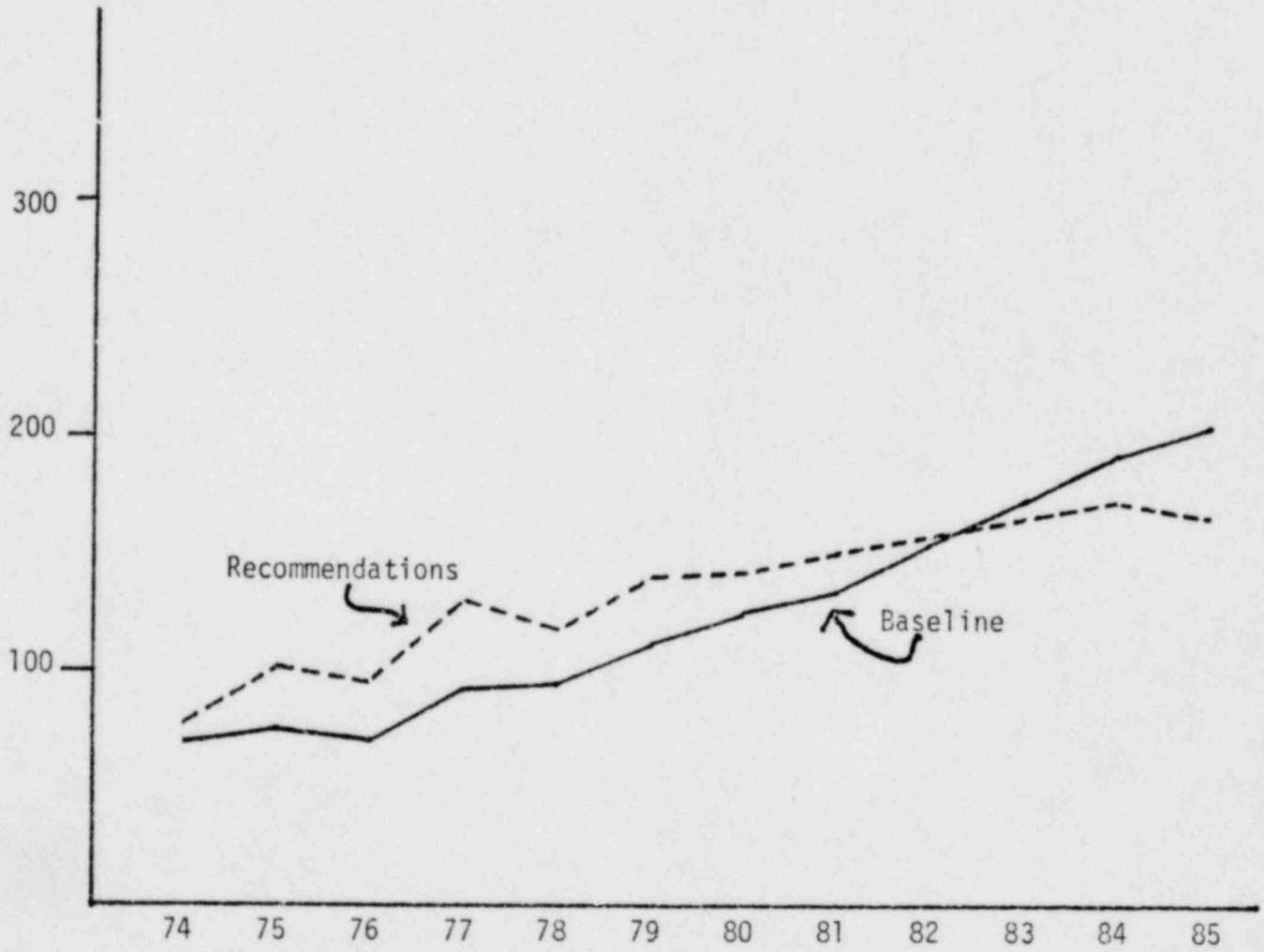


Figure 4

IMPACT OF RECOMMENDATIONS

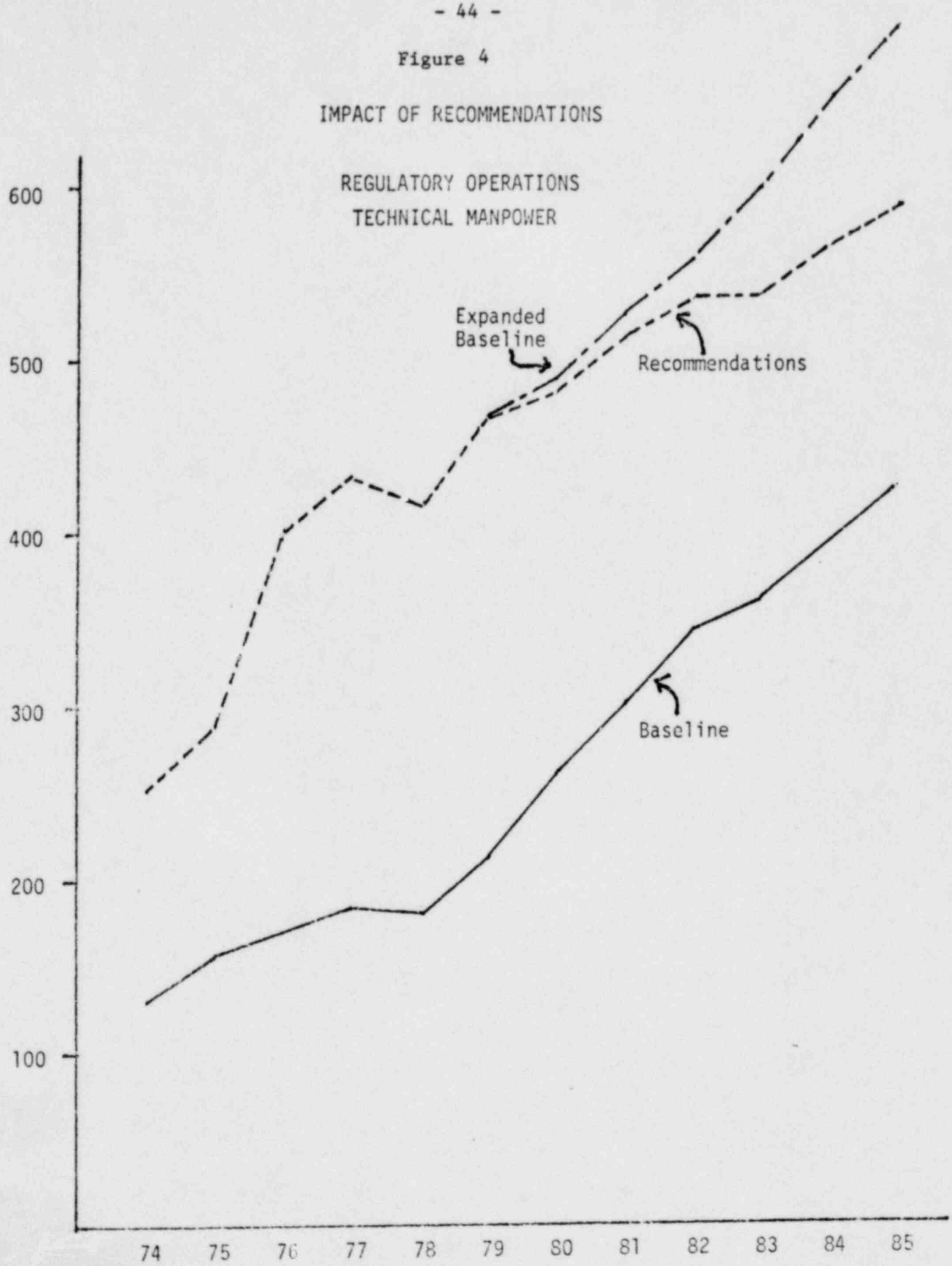
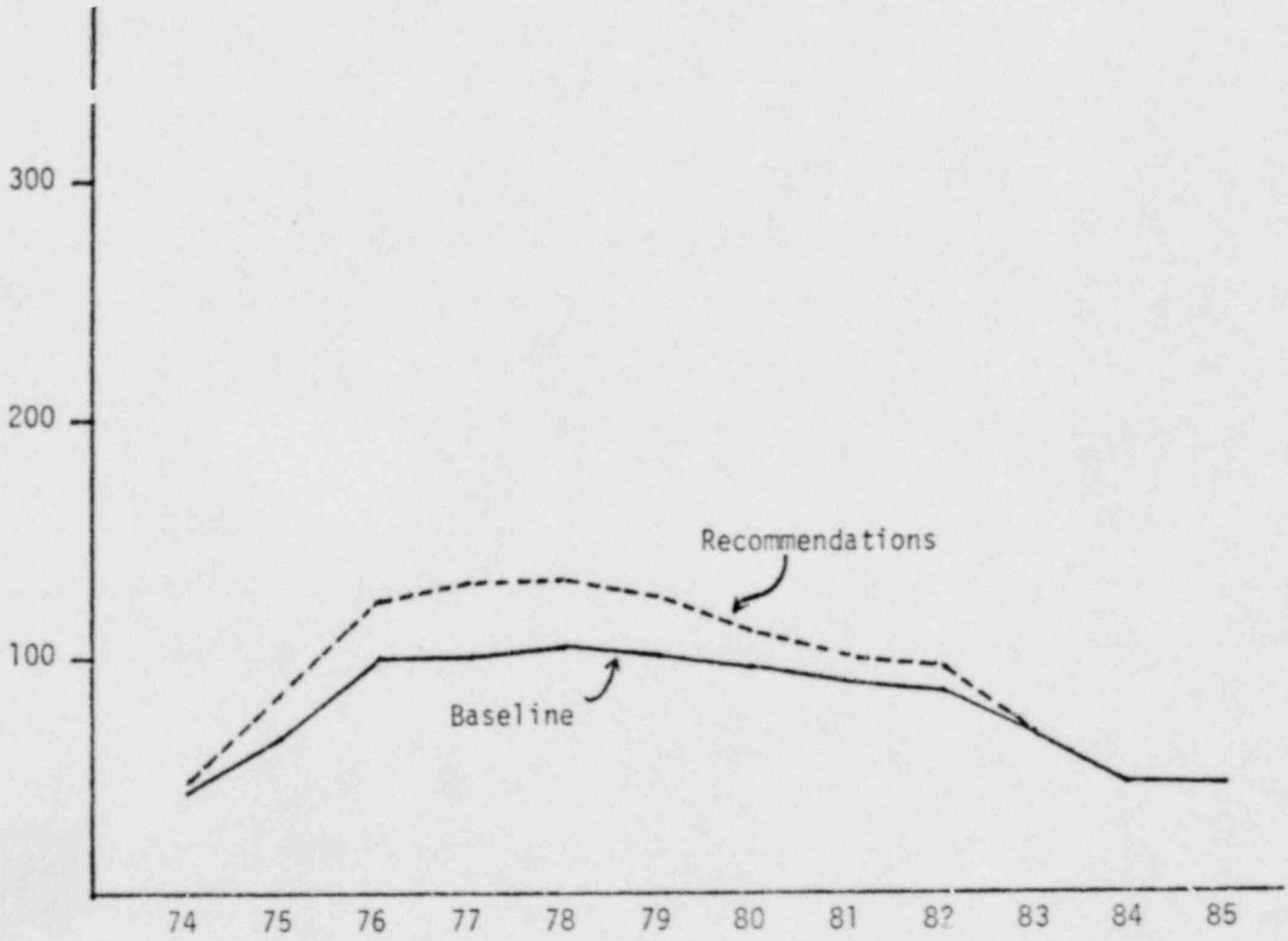


Figure 5

IMPACT OF RECOMMENDATIONS

REGULATORY STANDARDS
TECHNICAL MANPOWER



especially during the early years of implementation. The origins of this increase are identified primarily in the discussions pertaining to Figures 2 and 4.

2. By 1979 the manpower requirements associated with the Task Force recommendations have stabilized and remain essentially constant thereafter. This is accomplished primarily by phaseout of the CP and OL custom review process and efficiencies in the inspection program which are attributable to the advent of standardization.

3. The manpower requirements associated with the Baseline and Expanded Baseline Projections continue to increase proportional to the number of reactors under construction and in operation, except as modified by some of the efficiencies identified in section G.5 of this report.

4. The Task Force recommendations, when compared to the Expanded Baseline Projection, will result in a significant savings in Regulation manpower after 1982, when the number of CP applications will be markedly higher than today.

For the additional manpower investment portrayed by the increment between the Baseline and Recommendation curves in Figure 1, the AEC will be the beneficiary of increased efficiencies in the reactor review process; ability to react more reasonably to the currently predicted workload over the next 10-15 years; ability to respond more rapidly to any future unexpected nuclear penetration into the energy market; increased scope and depth of design review; and an increased site, NSSS, architect-engineer and vendor inspection program. The net result of all of these improvements should be a significant increase in the safety of reactors; and it will most certainly improve confidence in the low level of risk of nuclear power reactors, thus contributing significantly to increased public acceptance and a resultant shortening of the overall licensing process.

Figure 2 shows the impact of the recommendations on Technical Review. The additional staffing requirements in the early years results from the dual review effort in effect during the transition from "custom" reviews to standardized plant and designated site applications, as well as the increase in scope of the safety review effort inherent in the concept of standardization. Phaseout of these custom review processes, combined with the other efficiencies of standardization and site designation, facilitates a slight decrease and stabilization of

Technical Review manpower subsequent to 1979. The increment between the Baseline and Expanded Baseline Projections is due to the necessity to devote additional review effort to line routings, plant layout, equipment location, and qualification and preoperational testing programs.

Figure 3 shows a relatively insignificant impact of the recommendations on Reactor Projects. A decrease in manpower requirements occurs during the latter years. This is essentially accounted for by the phaseout of "custom" reviews.

Figure 4 shows the impact of the recommendations on Regulatory Operations. Of significance in this graph is the relative parallelism of the curves. In the cases of TR and RP, the recommendations in effect require the abandonment of the baseline procedures and substitution of a new way of doing business. In the case of Regulatory Operations, however, the recommendations have the effect of continuing present operations under essentially the same conditions assumed for the baseline, with newly established inspection efforts directed at NSSS suppliers, architect-engineers, and component manufacturers, as well as an increased scope of inspection efforts during the critical periods of construction and preoperational testing. The trend toward convergence of the curves after 1978 is the result of increased

efficiencies in the inspection process brought about by standardization of plant designs and a relatively stable number of utilities involved in comparison to the increasing number of plants, with concomitantly less inspection effort required per reactor.

Figure 5 shows the relatively insignificant impact the recommendations have on Regulatory Standards. The slight increase in manpower initially required is equally divided between the necessity to provide guidance and establish standards in the areas of safety review, standardization, and designated sites.

I. Impact of Greater Nuclear Penetration

Section H of this report details the manpower requirements (Baseline, Expanded Baseline, and Recommended) to cope with the workload projected by WASH 1139(72). Data from the 1973 AIF report "Resource Needs for Nuclear Power Growth", provides information as to the nuclear power growth that would have to occur, assuming 100% penetration of nuclear power into the electricity generating market in the next few years. This expanded penetration of nuclear power, if it were to occur, would require the review of 58% more CP and OL applications from 1975 thru 1985.

Figure 6 shows the end of year professional staffing levels that would be required, if the Regulatory staff were required to review the number of applications that would be submitted under the expanded power case, assuming 1.8 units per application and 1125 MWe per unit. Figure 6 also includes the necessary professional manpower to inspect the numbers of reactors under construction and in operation represented by this expanded power assumption. Three curves are provided (Baseline, Expanded Baseline, and Recommended) so that direct comparisons can be made to Figure 1.

Through 1985, a total of 24% more professional manpower would be required to handle the increased Regulatory workload, using the Task Force's recommendations (the Recommended Curves). This calculation assumes that the review of all "additional" applications would be handled under the designated site-reference design concept, and inspection by RO would be performed using all of the same assumptions as outlined in Section H.

J. Variations from Recommendations

1. General

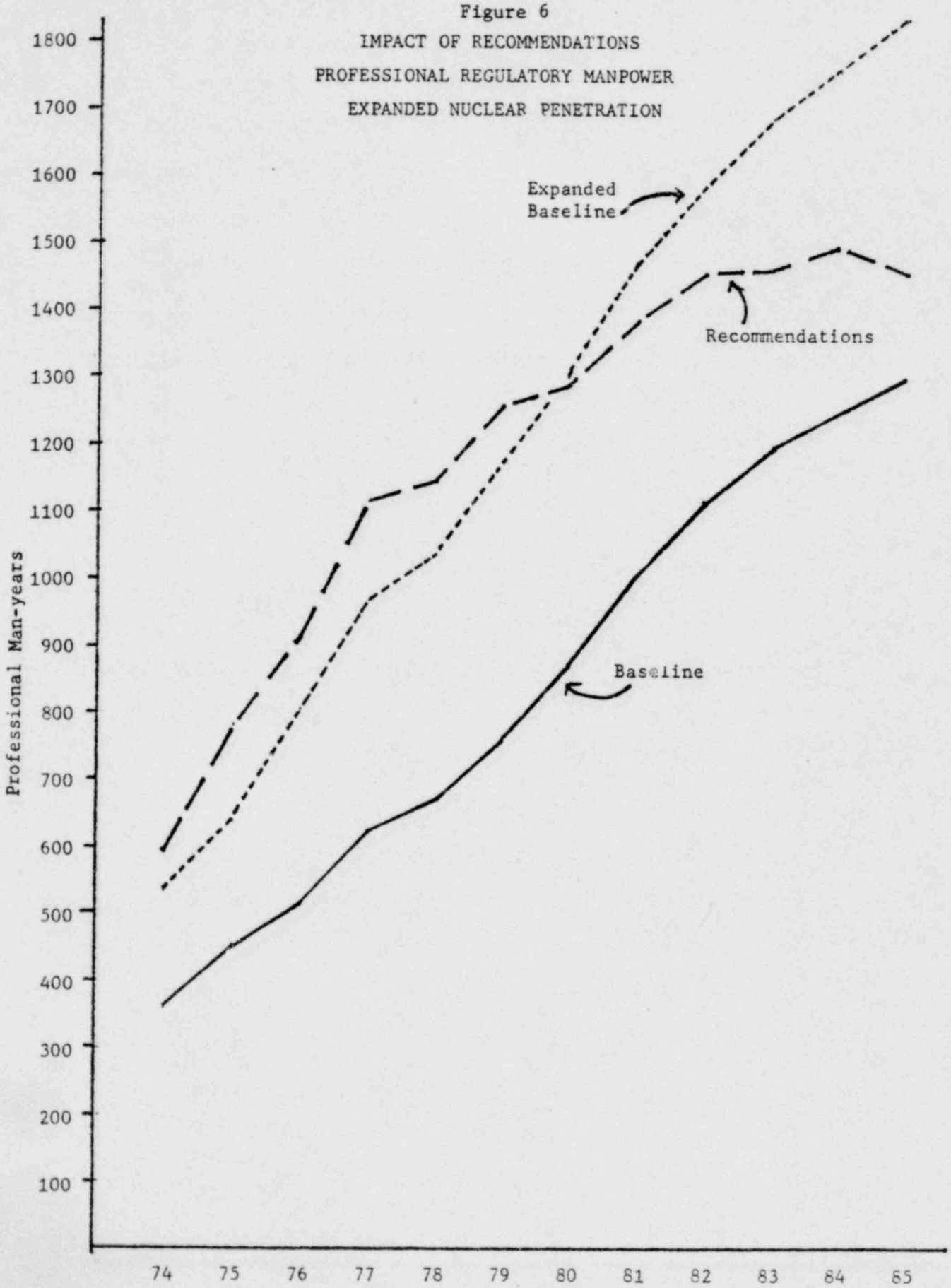
The information provided in the text of this report and in reference A clearly indicates that there are innumerable combinations of options and suboptions that could be considered

Figure 6

IMPACT OF RECOMMENDATIONS

PROFESSIONAL REGULATORY MANPOWER

EXPANDED NUCLEAR PENETRATION



in the selection of the proper course for Regulation over the next decade. It would be beyond the scope or resources of this particular Task Force to describe and estimate the budget for all (or even for a few) of these possible permutations. Recognizing that other specific options can be priced out from the information provided in this report, if needed, the discussion of variations that have a budget impact will be rather concise and qualitative.

2. Designated Sites

Regarding designated sites, the real advantages of this concept are:

a. greater assurance that the plant will proceed through any required Regulatory review and hearing process in a timely fashion (since all site aspects would be dispensed with prior to the CP application),

b. a decrease in the time from CP application to completion of the staff review process (this would be a small savings, unless one also assumes reference to a "standardized" design which has already been "AEC approved"), and

c. the establishment of siting criteria will act as a possible forcing function for standardization.

The budget impacts for designated sites only involve expediting the establishment of appropriate standards, plus the additional licensing review effort required to establish a backlog of designated sites. There should, however, be some compensatory manpower savings in elimination of the need to go through the site review process twice (at both the CP and OL stages).

Considering the rather small budgetary impacts as compared to the benefits, it does not appear fruitful to discuss significant variations from the proposed concept of site designation. To delay the process would not appear to be rewarding, since the increased budgetary requirements are on a one-shot basis, could be justified as a separate program in FY '76 - '77 that is consistent with the Administration's powerplant siting legislation, and would occur at a time when a temporary decrease in the CP application rate is anticipated.

3. Standardization

The concept of standardization combined with a necessarily expanded scope of review has a significant impact on budget projections. The recommendations of the Task Force will cost an average of approximately 90 additional professional man-years/year for FY '76 - '83 inclusive, as compared to no standardization and

our current way of doing business. By this time the efficiencies of standardization will be sufficiently realized so as to reduce the manpower required compared to our current way of doing business. It must be recognized that the resultant long term efficiencies for the staff and particularly for the industry, if a forceful standardization policy is pursued by the AEC, should make the short-term costs to Regulation and to the industry pale into insignificance.

With the above in mind, and in view of the professed energy crisis, it is difficult to envision acceptable variations, but a few are listed below.

a. The manpower expended on each standardized review could be decreased. This would be, in the judgment of the Task Force, a grave error which could significantly affect the aggregate level of risk for the projected large numbers of reactors in the future.

b. Standardization could be pursued as an option available to the industry. In the judgment of the Task Force, this would likewise be an error. Real standardization will probably evolve slowly if left to the natural forces of the marketplace, in spite of the tangible overall benefits to be accrued. A delay in implementation of standardization

would only mean that the AEC and the industry would not be ready for the surge of nuclear orders expected to start around 1979, and would certainly not be ready for any unanticipated increase in nuclear penetration of the energy market.

c. The number of standardized applications each year could be limited to 2 or 3, or, in the extreme case, only one U. S. Nuclear Plant could be permitted.. This would be more efficient from the standpoint of the AEC and the industry but would have severe effects on the economics of the nuclear industry.

4. Verification of Quality

The area of verification of quality (by inspection) represents the largest budgetary increase recommended by the Task Force; an average of approximately 190 additional professionals from FY '76 through FY '85 (decreasing to a low of approximately 135 additional professionals in FY '85). It is recognized that tangible returns to the AEC or the industry as a result of increased inspections are difficult to quantify. Thus, this most important function, which pertains directly to assurance of reactor safety, is more difficult to justify.

The Task Force heartily urges that any variations in this area be approached with caution, since the potential casualty would be that somewhat elusive value judgement called "level of risk." Possible variations are discussed below.

a. One could decrease the recommended inspection effort for NSSS's, A-E;s, and vendors. Since these are areas in which there is currently virtually no routine AEC inspection program, this variation seems plausible. However, the Task Force would recommend against this, since there have been significant deficiencies noted in the limited inspections performed, and there have been significant deficiencies in vendor produced products which have caused incidents. It must also be remembered that the majority of the nuclear power plant is fabricated offsite, and the quality of such activities must be verified if we are to fulfill our regulatory responsibilities.

b. One could require more licensee or third party inspection and less AEC inspection. This would be good and should be investigated further, but third party inspection capabilities are limited at present, and licensee inspections have not had a good track record. Also, both techniques would require substantial AEC monitoring with a resultant lessening in the savings of manpower.

c. One could reduce the number of vendors included in the vendor inspection program. While this is a possibility, it would be difficult to justify not routinely inspecting all manufacturers of major safety related equipment. In fact, once a good inspection base has been obtained it may be difficult to justify not expanding the program to include more manufacturers of minor safety related equipment and to parts and materials suppliers.

d. One could reduce the recommended expansion of the site-construction inspection. Considering the variability and complexity of site construction, the problems encountered in the past, the incidents that have occurred, and the areas to be included (as recommended by the Task Force) where inspections currently are not made at all, it is felt that this would not be a wise choice.

e. One could delay the manpower commitments. This would buy some time, but the budget problem would be even more difficult to solve later. It could be said that Regulation accepted potentially higher level of risk for several hundred plants, therefore why not do likewise for a few hundred more. This is truly the benefit-cost question.

5. Summary

In all of the "variations" considered above, the variations have been such as to reduce the level of recommended effort, or to delay it so as to reduce the near-term budget problems. While this is an understandable budget reaction, and it is frequently necessary to absorb cuts that are unwanted, it should be made clear that the Task Force's recommendations really represent the minimum program that is believed to be consistent with providing reasonable assurance that an appropriate level of risk will be achieved. If the Task Force felt constrainedtto be conservative in its approach (e.g., AEC certification of all safety related components), much higher budget numbers would result. In other words, an equal number of defensible "variations" could easily be prepared which would require a much greater manpower expenditure; both short term and long term.

It should be recognized that the Task Force's recommendations do not change the basic Regulatory philosophy; i.e., Regulation would still place the primary burden of performance on the licensee. There are many variations which could be prepared and defended which would imply shifting much of this burden to the AEC, however, this would cost at least ten times more than the recommended budget level. These approaches have their merits, and it is believed that they could perhaps improve the level of risk.

However, it is also concluded by the Task Force that such a major change in the role of the AEC would be difficult, if not impossible, to codify and implement at this time; and the margin of return might well be too small to justify the resource expenditure.

Quality Verification

A. Introduction

Quality verification is the process of testing, examination, or inspection which establishes or confirms the authenticity of a characteristic property or attribute with respect to excellence or grade of excellence when compared against a predetermined standard. Quality verification may be accomplished by several varied methods which will provide varying degrees of confirmation that a prescribed property or level of excellence has been attained. As espoused by the Atomic Energy Commission in fulfilling its regulatory functions, the design and operation of structures, components, equipments and systems, and the performance of personnel associated with the construction and operation of a nuclear power plant, establishes the required scope of quality verification.

Within Regulation, the Directorate of Licensing establishes the attributes which must be verified, the Directorate of Regulatory Standards develops the standards against which the excellence of the prescribed attributes are compared, and the Directorate of Regulatory Operations determines whether the quality of the items within the AEC of quality verification is as presented by the licensee in his lication. Quality verification, as envisioned by Regulation, is required to provide adequate confidence that:

1. Regulatory requirements and design bases related to nuclear power plants, as defined in paragraph 50.2(u) of 10 CFR and as specified in the license application, are correctly translated into specifications, drawings, procedures, and instructions, and
2. Components and equipment fabricated and tested in manufacturer's facilities conform to the specifications, drawings, procedures, and instructions, and
3. Structures, systems, and components constructed, erected and tested at the nuclear power plant site conform to the specifications, drawings, procedures, and instructions, and
4. Activities such as operating, testing, repairing, maintaining, and modifying nuclear power plants are conducted in such a manner as to assure safe operation.

B. Levels of Quality Verification

1. Quality Assurance Program

One of the basic ingredients in quality verification is the overall programmatic control employed by the licensee. This is usually referred to as the quality assurance program and is established by the applicant and/or his contractors to control and implement the commitments made in the application (Safety Analysis Report), the requirements of his license, and the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants."

The quality assurance program is required to be documented by written policies, procedures, or instructions and is to be implemented throughout the life of the plant as prescribed by this documentation. Structures, systems, components, and organizations (including their functions) included in the program are required to be specified. Methods of controlling activities affecting the quality of the identified structures, systems, and components, consistent with their importance to safety, are also to be delineated. This includes control of design; environmental conditions; use of special processes, test equipment, tools and skills; and verification of quality by means such as inspection and test. The QA program must also provide for the training of personnel performing activities which affect quality. The status of the QA program is to be periodically audited and reviewed by the licensee or his delegate to ascertain its adequacy.

The QA program, as examined and evaluated by Regulation, must provide the required confidence to the AEC that the licensee or his delegate has developed a program of quality verification consistent with the requirements of 10 CFR, Appendix B, and the application. By itself, however, this examination will not provide confidence that the licensee has implemented the program.

2. Quality Assurance Program Implementation

As discussed in paragraph B.1, examination of a QA program by the AEC does not provide assurance that the QA program has, in fact, been implemented. Such assurance is necessarily obtained through inspection of the design, fabrication, erection, testing, training, and other action criteria as implemented at the plant site, the offices of the nuclear steam system supplier and architect-engineer, and component manufacturers. Inspection of QA implementation may be accomplished through physical examination of the process steps as they are performed, coupled with a review of the attendant process documentation. Although the degree of confidence placed in a quality verification program such as this depends upon the overall inspection effort by the regulatory agency, a well devised sampling/audit program should provide reasonable assurance that the quality assurance program has been implemented consistent with regulatory requirements.

3. Design Verification

The design bases as defined in paragraph 50.2(u) of 10 CFR and the AEC design criteria as delineated in 10 CFR 50, Appendix A, "General Design Criteria" are utilized by the licensee and his contractors in establishing the nuclear power plant design presented in the application. The licensee, as part of his

quality assurance program, is required to develop controls to provide for the correct translation of design requirements into specifications, drawings, procedures, and instructions.

The design of the nuclear power plant, as presented in the application, is reviewed by the Directorate of Licensing to the extent deemed necessary to provide the required confidence that the elements of design important to safety have been considered, the design should provide an acceptable level of risk, and the specified reliability should be achieved. This is contingent upon the designated design standards and specifications involving the manufacture, fabrication, installation, erection, testing, and/or operation of the safety related components, equipment, and systems comprising the nuclear power plant being implemented consistent with regulatory requirements. Design verification may be accomplished through qualification testing of the completed component, equipment, or system against the specified design criteria. (This is similar to prototype testing whereby a component is operated through its expected life cycle and under extreme conditions.) For example, a containment spray pump may be run through 1000 start/stop cycles and then run for 60 hours under post-accident conditions, if that is the design performance criteria. Design

verification may also be accomplished through calculational techniques. The Directorate of Regulatory Operations confirms through a sampling inspection program that the licensee has conformed with the requirements of the designated design standards and specifications.

4. Product Quality Verification

Product quality verification refers to the process of inspection, examination, or testing of components, equipment, systems, or structures whether fabricated at the manufacturer's shop or erected at the nuclear power plant site. Typical end products in the two categories would be a safety injection pump and the containment structure. The quality verification process associated with these and other products installed in the nuclear power plant should be consistent with the products' impact on safety. The scope and depth of the verification process, as related to specific components, is thus quite varied. The basic elements of the process should include measures to assure that:

a. the required design, testing, and quality assurance specifications are accurately reflected in purchase documents.

b. the product, including any subcomponents, is fabricated, manufactured, installed, tested, and/or erected consistent with the purchase document requirements.

c. the product is operationally tested to the degree necessary to demonstrate operability consistent with design application. This verification may consist of tests conducted either in the manufacturer's shop or at the nuclear power plant site and is in addition to any design proof (qualification) testing discussed in paragraph B.3.

The program used by a regulatory body to verify product quality may range from placing total responsibility on the licensee to complete independent verification of the product's attributes.

C. Quality Verification Programs of Other Agencies

Several other agencies, both regulatory and nonregulatory in nature, have established quality verification programs of varied scope and depth. Many of these agencies issue a certificate, and thus their quality verification program is thought of as a certification process. The certificate, in general, attests to the successful completion of a quality verification process (by the product supplier and/or by the certifying body). The Task Force reviewed the quality verification programs employed by the Federal Aviation Administration (FAA), Food and Drug Administration (FDA), Underwriters Laboratory (UL), American Society of Mechanical Engineers/National Board of Boiler and Pressure Vessel Inspectors (ASME/NB), and the AEC Nuclear Weapons Program (WEAPONS). Table I provides a comparison of several key quality verification parameters associated with the five agencies during initial certification and periodic followup review.

TABLE I

Type of Inspection	Organization	Product Certified	Standards/ Established by	Inspection Conducted by	Cost of Insp Borne by
Initial (Precertificate) Periodic Followup	Underwriters Laboratory (UL)	Electrical, Heating, Air Conditioning, Refrigeration, Burgulary, Fire Protection & similar equipment	UL establishes requirements in "Standard for Safety" UL establishes scope of reinspection in "Followup Service Procedure"	UL Personnel[1/] Manufacturer and UL personnel	Mfgr.
Initial (Precertification) Followup	American Society of Mechanical Engineers/ Nat'l. Board of Boiler & Pressure Vessel Inspectors (ASME/NB)	Pressure vessels and other pressure retaining components - only pressure retaining capability is certified	ASME establishes requirements in the Boiler and Pressure Vessel Code (Section III-Nuclear)[2/]	Survey Board consisting of 4-5 personnel representing ASME, NB, Inspection Agency (IA) and state. IA such as an insurance company or state. No firm inspection criteria.	Mfgr.
Initial (Precertification) Followup	Federal Aviation Administration (FAA)	Aircraft, engines, propellers, and related components	FAA establishes detailed standards specifying technical requirements	FAA personnel and manufacturing personnel certified by the FAA as FAA in-plant representatives[3/]	FAA/Mfgr.
Initial Followup	Food and Drug Administration (FDA)[4/]	Investigational drugs - includes human testing (No sales allowed) New drugs - sales allowed	IND[5/] No specific standards established by the FDA. NDA[6/] FDA and their representatives review submittals by applicant. Decisions based on professional judgment.	FDA conducts review of IND submittal FDA conducts review of NDA submittal and performs inspection of documentation and manufacturing facility.	FDA
Initial Followup	Nuclear Weapons Program (AEC/ALO)[7/]	Weapon components and complete warhead	Specific requirements delineated in purchase orders prepared by procuring activity (AEC/DOD)	AEC resident inspectors verify manufacturer quality assurance performance	Mfgr. and AEC/ALO

Type of Inspection	Organization	Inspector Qualification	Source of Authority	Certificate Issued by	Bases for Certificate Withdrawal
Initial (Precertificate)	Underwriters Laboratory	As determined by UL	Manufacturer's request for Certification	UL certifies right to label as "UL approved"	- NA -
Periodic Followup	(UL)	As determined by manufacturer and UL as appropriate	UL agreement with mfgr. for retention of certification	Product labelled by manufacturer	Failure to pass periodic UL inspection at manufacturer's plant and UL "in-house" tests using "Followup Service Procedure"
Initial (Precertification)	American Society of Mechanical Engineers/ Nat'l. Board of Boiler & Pressure Vessel Inspectors (ASME/NB)	As determined by ASME/Nb - detailed requirements not specified	Boiler and Pressure Vessel Code - In some states it is a legal requirement	ASME issues "N" stamp to manufacturer for use on products	- NA -
Followup		Established by N.B. Inspectors qualified on basis of a test		Manufacturer and IA for individual products	Code violations as reported by IA and confirmed through investigation by ASME/NB
Initial (Precertification)	Federal Aviation Administration (FAA)	FAA inspector requirements established by FAA. Designee qualifications established by FAA and mfgr. Minimum qualifications are delineated.	FAA regulations upon receipt of manufacturer's request for certification	Type, production and airworthiness certificates issued by FAA	- NA -
Followup			As specified by conditions of certificate	- NA -	Violation of FAA regulations or conditions of certificate
Initial	Food and Drug Administration (FDA)[4/]	FDA personnel as established by FDA. Investigators as proposed by drug sponsor.	FDA regulations upon receipt of IND from drug sponsor	FDA	Violation of commitments specified in IND or adverse reports from drug sponsor
Followup			FDA regulations upon receipt of NDA from drug mfgr.	FDA	Violation of commitments specified in NDA or adverse reports from the medical community.
Initial	Nuclear Weapons Program (AEC/ALO)[7/]	Well defined by AEC/ALO	Purchase contract specifies contractor and subcontractor requirements	Mfgr.	
Followup					Inspections or tests by AEC inspector indicate breakdown in manufacturer's QA program.

NOTES

1. Appeals from UL actions which withhold "listing" (certification) may be reviewed by the National Bureau of Standards. UL reports which lead to "listing" are public. AEC Regulatory Guide 1.52 identifies some UL standards as being an acceptable method of compliance with certain regulatory requirements.

2. The ASME code is applicable to the manufacturers of pressure vessels, parts and appurtenances, valves, and the site installer. ASME reports which lead to "authorization" are not public. The ASME code is presently accepted by less than 35 states.

3. FAA has established detailed technical standards to be used in determination of product quality. Manufacturer reinspected by FAA personnel every 18 months unless inspected earlier due to cause.

4. FDA also has an antibiotic and insulin certification program covering batch certification as to quality. Testing is supported by fees from manufacturers and accomplished by FDA.

5. IND (Notice of Claimed Investigational Exemption for a New Drug) is a detailed proposal covering drug manufacturing procedures, pertinent prior testing, test plan, qualifications of test investigators and report requirements. IND includes sponsors' claims for drug and methods to prove safety and efficacy. FDA regulations give only broad scope requirements.

6. NDA (New Drug Application) contains results of IND tests, details of production process and personnel qualifications of personnel responsible for quality. FDA regulations give only broad scope requirements.

7. AEC/ALO requires that manufacturer's establish and implement an adequate QA program. The AEC/ALO inspectors verify same through inspection techniques including independent samples and testing.

The quality verification programs of the various agencies are in many ways similar to the quality verification program recommended by the Task Force. The primary differences are prompted by the recognition of regulatory versus nonregulatory functions, size of components involved, complexity of the overall product requiring verification, and desired level of risk. Several significant aspects of the FAA and Nuclear Weapons quality verification systems are discussed herewith for comparison to the Task Force recommendations.

1. Federal Aviation Administration

The FAA issues three certificates: Type, Production, and Airworthiness. A Type Certificate attests that the proposed complete aircraft design and construction are consistent with FAA requirements. The review process associated with a Type Certificate is performed primarily by FAA Regional and Headquarters personnel and to a limited extent by FAA designees (DER) (manufacturer's personnel who perform an engineering and evaluation function for the manufacturer and are authorized by the FAA to represent the FAA in certain areas). The review process takes from 3-5 years and involves FAA or DER review of all engineering design and performance criteria and all final design drawings. Prototype products are verified by the FAA and DER to conform with the design and performance criteria. This includes prototype testing of the complete aircraft. To accomplish this effort the FAA establishes

detailed standards providing numerical indications of the safety factors to be used in the design, configuration details, and engineering formula.

They are more detailed than many of the AEC Regulatory Guides. In the review process the FAA does not perform a detailed checkout of most calculations; however, the end results are reviewed for reasonableness.

The FAA issues a Production Certificate governing the manufacture of the specific product, based on inspection of a manufacturer's QA program which demonstrates his capability to manufacture an aircraft, engine, or propeller consistent with the requirements of the Type Certificate. The manufacturer is held responsible by the FAA for quality assurance through all tiers of material and component suppliers. The suppliers, in general, are not inspected by the FAA or designees. Approximately every eighteen months, the FAA performs a Quality Assurance Systems Analysis Review (QASAR) of the manufacturer's operation. The QASAR, performed by a three-five man FAA inspection team over a two-week period is designed to evaluate the overall effectiveness of the manufacturer's QA effort.

During the production phase, the DMIR's (manufacturer's personnel who normally perform a quality control supervisory function for the

manufacturer and are nominated by the manufacturer and authorized by the FAA to represent the FAA) perform most of the day to day inspection effort. This includes: verification that the manufacturer's stated quality assurance program is being implemented, witnessing of selected tests, and review of equipment and data. Also, DER's review detailed design drawings and approve design data regarding certain types of design changes. In addition, once a Type Certificate has been granted, the FAA assigns an FAA Principal Inspector to monitor the overall inspection activities at the manufacturing facility. He is responsible for assuring that the Federal Aviation Regulations and requirements of the Type and Production Certificates are complied with and that the QASAR inspection is performed at the prescribed frequency. In general, he only audits the performance of the FAA designee functions.

The Airworthiness Certificate is issued by the DMIR based on the manufacturer's satisfactory completion of the checks and tests delineated in the Type and Production Certificates. Each aircraft rolling off the production line receives an Airworthiness Certificate.

2. Comparison of FAA and AEC Quality Verification Programs

The recommended AEC quality verification effort is comparable to the FAA program in many respects. Type Certification is the FAA version of the current AEC, CP and OL application review process. Whereas the

FAA reviews all design aspects of the product, the AEC reviews only those deemed to be safety related. Upon full implementation of the recommended standardized plant concept, Type Certification will be essentially equivalent to the nuclear plant safety complex final design review process, considering the fact that the AEC will issue some form of "certification" once the design has been reviewed and is deemed acceptable.

The FAA holds the manufacturer ("certificate holder") responsible for assuring the quality of components received from suppliers. The manufacturer may verify quality through review of the supplier QA program or testing of the component. The FAA does not inspect component suppliers (unless they hold FAA certificates), except upon the request of the manufacturer's FAA inspector. The recommended AEC component quality verification program similarly holds the licensee (utility licensee-standardized plant "certificate holder") responsible for assuring the quality of components. In addition, the recommended AEC quality verification effort will verify the quality assurance program of major vendors and the adequacy of these programs and of the licensee's efforts, through a sampling inspection program at the major component supplier's shops. The recommended AEC program will also require that suppliers certify the quality of the products to the purchaser and that appropriate qualification testing of specified products be performed.

Prior to issuance of a Production Certificate the FAA conducts a quality assurance program inspection of the manufacturer. This is similar to the current AEC pre-docketing and pre-CP quality assurance inspections.

After receipt of the Type and Production Certificates all major design changes must be approved by the FAA. The DER's, in their review functions, determine whether all design changes have received the required approvals. This is similar to the AEC requirements regarding PSAR and FSAR amendments, and 10 CFR 50.59 requirements. The AEC determination as to whether design changes have received suitable approval is based on periodic inspections (on a sampling basis) by Regulatory Operations and the attendant finding as to whether the plant was constructed or modified consistent with the application.

The DMIR's and DER's, as resident personnel, are more familiar with the everyday activities and changes at the manufacturing plant than are the RO inspectors r Licensing reviewers, and hence should be able to verify more absolutely the appropriateness of design changes and of production in accordance with design. On the other hand, they are manufacturer's employees, recommended by the manufacturer, and designated by the FAA to uphold the FAA interests, which has the potential to lead to a conflict of interest problem. The advent of a resident inspection program on the part of the AEC should increase

our confidence level that the plant is constructed consistent with design, but it would have to be carefully controlled and audited.

Periodically during aircraft production the FAA conducts a detailed QASAR inspection. This is comparable to the recently completed RO management inspections at operating reactors, which the Task Force recommends be continued.

DMIR issuance of the Airworthiness Certificate on behalf of the FAA is contingent upon quality verification activities similar to those addressed by the AEC in issuance of an Operating License. Whereas the FAA-required prototype testing program is completed by the manufacturer prior to receipt of a Type Certificate, the AEC-required prototype testing program is rather minimal in nature, and therefore much reliance is placed on the preoperational and operational test program established for each individual plant. This type of testing is probably more comparable to the airworthiness testing program. The Task Force recommends more emphasis be placed on qualification (prototype) testing.

In summary, the FAA and recommended AEC quality verification programs are quite similar even to the point of possible Certification (Licensing). Table II provides, in summary form, a brief comparison of the FAA system to the recommended AEC system.

TABLE II

COMPARISON OF FAA/AEC QUALITY VERIFICATION PROGRAM

	FAA	AEC
• Conditions for Application	FAR's	10 CFR 50 and Regulatory guides
• Approval and Inspection Process		
• Phase I	Type Certificate	Final Standardized Design Certificate
• Length of Time, Phase I	3-5 years	4 years (design and approval)
• Level of Technical Review	Compliance with FAR's. Approve all eng. design, performance criteria, and final drawings	Compliance with standards and rules. Approve engineering design, performance criteria, selected final drawings, plant layout, and line routings
• Quality Assurance Inspection	QASAR inspection program	Pre-docketing, NSSS, and AE QA inspections
• Review Organizations	FAA Regional and HQ offices and FAA approved designees.	AEC HQ (Licensing and RO) and RO Field Offices (QA implementation)
• Standards	Appropriate mil'stds, FAR's, company standards, etc.	Applicable codes and standards, AEC rules, and Regulatory guides.
• Phase II	Production Certificates and production	Construction Permit and construction
• Length of Time, Phase II	Approximately 6 months for Certificate (Varies with product)	Approximately 6 months for the CP
• Responsibility and Level QA control	Certificate Holder or Manufacturer, thru all tiers of mat'l and component suppliers	Utility applicant and Design Certificate holder, thru all tiers of mat'l and component suppliers

TABLE II (Cont'd.)

	FAA	AEC
• Daily enforcement of QA at prime manufacturer's shop (construction site)	Certificate holder, FAA inspector, approved FAA designees (after issuance of Production Certificate)	Applicant, Certificate holder, and Resident and Regional based AEC inspectors
• Enforcement of QA at major component vendors	Certificate holders. FAA inspector or designees, if Production Certificate is held. No FAA inspection if Production Certificate is not held, unless requested by FAA inspector of Certificate Holder	Applicant and Design Certificate holder. Routine QA implementation and selective hardware inspection by AEC inspectors
• Enforcement of QA at minor component parts, and materials vendors	Certificate holders or purchaser (Essentially no FAA inspection)	Applicant, Design Certificate holder, or purchaser (Very little routine inspection by AEC, except spot checks and problem oriented)
• Methods of overall QA program evaluation	QASAR of prime manufacturer and major subcontractors who hold Production Certificates	QA inspection of applicant, NSSS, AE, and all major component vendors
• Phase III	Airworthiness Cert.	Issuance of OL
• Level of Inspection	Preoperational and flight testing inspections by FAA or designees	Preoperational and startup testing by AEC resident and Regional personnel

3. Nuclear Weapon Program

In the nuclear weapons program it is the responsibility of the AEC (General Manager) to design the product to meet a specified need, evaluate the product from the standpoint of safety, assure the quality of the product, test the product, and assess product reliability.

In the design of the weapon, the design laboratories under contract to the AEC utilize detailed logic analysis including intuitive analysis, component failure and product effect analysis, and fault tree analysis, coupled with known component performance data, to assess the overall design reliability and level of risk. In this context, minor changes in product design are not allowed unless safety and reliability are measurably increased.

The suppliers of the major components and the manufacturer who assembles the weapon are required by the AEC to have an effective quality control program. The component suppliers and the manufacturer certify to the AEC that the product meets all specification requirements. The AEC does not specifically delineate the requirements of the contractor's certification program; however, the AEC has developed a rigid demerit system for rejection of the components. Consequently, to preclude loss of revenue, the contractors have developed good certification systems which include the key elements of quality verification (detailed standards, procedures, testing, and evaluation).

The AEC does not perform a certification of the contractors' product or performance. To the contrary, the contractors certify product quality to the AEC. The AEC evaluates the validity of the contractors' certification through the AEC QA program consisting of verification inspections, audits, and quality evaluations.

Verification inspections are performed by AEC resident inspectors at the contractors' facilities. The AEC inspector has independent capabilities to perform the checks necessary to verify the quality of the products supplied to and by the contractor. The checks are performed in accordance with detailed inspection procedures developed under AEC contract by Sandia Laboratory. These checks are conducted on each component, or on a sample basis, depending upon the component relation to safety, production rate, and past performance history.

Approximately every two years, in-depth audits of the QA programs employed by the contractors are performed by the AEC. In addition, special audits of problem areas are performed as appropriate.

Quality evaluations, in the form of comprehensive destructive and nondestructive examinations of small quantities of important components, are conducted by the AEC independent of the contractors' examinations. These evaluations are performed on both production and stockpile weapons on a sampling basis.

4. Comparison of AEC Weapons and Reactor Quality Verification Programs

The AEC weapons and the recommended AEC reactor quality verification programs are similar in scope but somewhat different in the depth of review and the techniques employed. These differences may be attributed to the role of the AEC in both programs (customer versus regulatory), and the size and complexity of the products involved.

In the weapons program the AEC has resident inspectors located in the contractors' shops. At the present time the AEC reactor program does not employ resident inspectors. In the recommended program, the principle of resident inspection may be employed at construction sites. It is recognized that the limited number of contractors in the weapons program (9), in conjunction with the relatively large number of inspectors (130), requires utilization of the resident inspector concept.

The AEC weapons quality verification program includes a great deal more independent qualification testing than will be accomplished in the inspection program recommended by the Task Force. The size of the components utilized in nuclear power plants would necessarily require a much greater expenditure of resources, if a comparable independent testing program were to be initiated in the reactor program. Recognizing that increased qualification testing requirements on the manufacturers, coupled with a significantly increased sampling testing program by AEC

certified laboratories, are recommended by the Task Force, it appears that a greater increase in the AEC independent testing program is not warranted at this time.

Like the weapons program, management inspections of overall QA programs and systems are employed by the AEC in the reactor program (predocketing, pre-CP, pre-OL, and management inspection of operating reactors).

A considerable amount of independent destructive and nondestructive examination is performed by the AEC in the weapons program. This concept has been utilized to a limited extent in the reactor program (environmental samples, radiological analyses); however, in the area of hardware testing, Regulation relies more heavily on witnessing the tests and reviewing the licensees' test results. This difference may be attributed, in general, to the AEC's role of customer in the weapons program and the size of the components in the reactor program. Increased design qualification testing and inspection of the component supplier shops are included in the recommended reactor inspection program. Also, the AEC-certified testing laboratories would perform independent tests of certain critical components, as prescribed by the AEC.

The AEC in the weapons program has collected more reliability data than has the AEC in the reactor program. The Task Force recommendations include increased effort devoted to the continued development

and implementation of a reliability information system to be used in assessment of level of risk.

Minor changes in product design are not allowed in the weapons program unless the level of risk or reliability may be favorably affected to a measurable degree. This concept is similar to the standardized plant design concept recommended by the Task Force.

In summary, the AEC weapons program is comparable to the AEC reactor program in the area of QA inspection; however, the weapons program devotes considerably more resources to product inspection, independent qualification and acceptance testing of components, and development of design. It must be remembered, however, that in one program (Weapons) the AEC is the customer and in the other (Reactor) the AEC is the regulator.

D. Certification in Perspective

Certification, as used by a regulatory body or inspection agency, is a written testament that a product or service meets certain required standards. The certificate thus conveys implied or specific warranties on behalf of the certificate holder. The actual verification effort by the certifying agency is not reflected in the certificate. Our reactor licensing process is essentially a certification. If we were to expand our certification program

to include component vendors, nuclear steam system suppliers, and architect engineers, the following items would first have to be considered:

1. Applicant

   At the present time, there are several hundred major vendors, 13 architect-engineers, and 5 nuclear steam system suppliers known to be actively engaged in supplying major safety related products or services utilized in the design and fabrication of components for nuclear power plants. It is conceivable that the number of lesser known vendors and subvendors engaged in supplying components or vying for a piece of the nuclear market is approaching a thousand, and suppliers of safety related materials and component parts could number in the thousands. Since possession of an AEC certificate, with its implied warranties, could provide an economic advantage, we should expect to be innundated with applications for certification. The magnitude of this problem could be reduced somewhat by limiting the number of applicants for any one type of component; however, a system of justifying the reasons for accepting one application versus another would then have to be established. Reviews of these "rejected" applications would have to be made, and there could be severe restraint of trade problems. Likewise, to reduce the number of applicants, the

types of components to be certified could be severely restricted. This might be hard to justify, however, since the technical basis for elimination of most safety related components would be difficult, especially if "certification" were being touted as the primary technique for improving the level of risk. Unless carried to extremes, neither of the foregoing options would provide the needed reduction in the number of applicants to avoid an impossible increase in workload. Another problem is that some of the major vendors actively engaged in the supply of safety related components to the nuclear market are foreign based. The Atomic Energy Act (Sec. 103.d) prohibits the licensing of persons or activities which are not under or within the jurisdiction of the United States. Unless legislative changes in the Act were implemented, certification of foreign vendors could not be achieved.

2. Standards and Specifications

One of the primary building blocks in the development of any quality verification program is the development of explicit standards and specifications defining the required specific attributes of the design, process technique, fabrication, manufacture, erection, testing, and operation of safety related components, equipment, systems, and structures associated with nuclear power plants. Such standards and specifications define

for the applicant the requirements which must be met, if a product is to be utilized in the construction or operation of a nuclear power plant. These standards and specifications also provide a fixed benchmark by which the regulatory safety review and field inspection processes can gauge the quality of the product under their purview.

Although every type of quality verification program is dependent upon the development of good standards and specifications, it is especially important if certification to, or by, the AEC is the end result of the verification process. The standards and specifications establish the attributes against which the products are publicly judged. At the present time such standards and specifications do not exist in sufficient detail for all safety related components to permit the establishment of a comprehensive component certification program.

3. Enforcement

Enforcement, as employed by a regulatory agency, has varied connotations and may be employed through various techniques ranging from discussion and pressure being put to bear via threats to the public image, through orders to suspend or revoke licenses. Enforcement actions employed should be consistent with the severity of the deficiency and be of such a nature to cause corrective actions to be promptly initiated.

Enforcement should also deter recurrence both on a specific and generic basis. Quality verification programs without AEC certification (licensing) do not permit sanctions of an official nature to be employed directly against the component supplier. As in our present inspection program, if a deficiency in a product is identified, the official enforcement action must be taken at the license holder (utility) level. Direct discussion and public image pressures may also be effectively used. Licensing of vendors, nuclear steam system suppliers, and architect-engineers would permit direct enforcement action and expand the scope of enforcement actions available to the AEC when dealing with these entities.

4. Inspection

Inspection, as performed by a regulatory agency, is the process by which the design, design implementation, and operation of a product is ascertained to be consistent with established standards and specifications. Inspection, like Standards, is a cornerstone in the establishment of any viable quality verification program. The inspection effort should be developed to confirm the various levels of quality verification. We may elect to verify the existence and implementation of effective QA programs, design concepts, detailed design, design implementation, fabrication, manufacture, erection, testing, operability, or any combination of the foregoing as associated with safety related products used

in a nuclear power plant. The required inspection effort will necessarily vary consistent with the nature of any quality verification program (i.e., the importance of the inspected activity to safety, the "track record" or previous experience with that activity, and whether the purpose of the inspection is to "assure" or just "provide reasonable assurance" of the quality of the inspected activity). Since certification implies a degree of warranty, we would expect to increase our inspection efforts significantly with certification (licensing).

5. Confidence Level

Confidence level, if it is to be expressed in absolute terms, must be tied to a measurable entity. If a safety review and inspection process verifies a certain number of possible action points within a sphere of activity without finding any deficiencies, we may conclude that we are confident (to some degree) that almost all the action points within the specific sphere of activity meet regulatory requirements. For example, if there are 800 welds in the primary coolant system and we perform a complete audit of the licensee's inspection results of 50 of the welds (including paper and process application review for all aspects of the weld requirements) and find no problems, we may be 90 percent confident that at least 99 percent of the inspected attributes as pertaining to the other 750 welds are acceptable. It is obvious that if

only AEC resources were considered.as applied to the various inspection facets of a nuclear power plant, (QA programs, implementation of QA programs, design review, fabrication, erection, testing and the numerous attributes within each activity sphere) the result would be an extremely low confidence level that the product will perform as designed. If the confidence level were based only on discrete measurable entities as examined by Regulation and were expressed in absolute terms, any attempt to raise the confidence level to any acceptable value would be prohibitive from a manpower standpoint and would tend to replace the inherent responsibility that the licensee (product supplier) should retain.

Confidence level, as presently used in Regulation, is generally expressed in terms of "reasonable assurance." As related to nuclear plant safety and operability, the AEC must have reasonable assurance that the nuclear power plant has been constructed consistent with Regulatory requirements and can be operated safely. This philosophy has been a cornerstone of our Regulatory policy. This, of course, is a qualitative technical judgment formulated by examination of the licensee and contractor quality assurance programs and implementation of same. Certification, per se, would not establish or change the level of confidence. It is the quality of the review, testing, and inspection functions

performed by the licensee and his contractors (or the AEC if we choose to change our philosophy of regulation) which establishes the confidence level. However, the AEC must have a comprehensive sampling/audit inspection program to verify the adequacy of the licensee's quality verification efforts.

6. Program Control

Quality verification programs, whether established by regulatory or nonregulatory agencies, share similar program control obstacles. The major problems encountered include maintenance of inspector objectivity, utilization of a common reference standard, uniform interpretation of requirements by the inspectors, and training. A regulatory agency has the additional problem of maintaining an even-handedness in administering enforcement of the regulatory requirements. The identified program control problems may be readily overcome through the development and implementation of recognized management tools such as audits, inspection and enforcement guides, and training programs.

Certification has no direct impact on the foregoing programmatic matters; however, it may pose an additional burden in the form of application processing and certificate issuance control.

7. Antitrust Considerations

Quality verification programs, including certification, should not pose any difficulty in relationship to antitrust matters,

provided any reduction in competition in the marketplace is based on clearly established performance requirements (safety or reliability) or needed improvements in efficiency or resource allocation. It is probable that there could be antitrust problems, if our requirements were established so as to "force" all but a few companies out of the nuclear field for no real reason except to limit the number of certificates issued. If, however, our requirements were established consistent with the fulfillment of our assigned responsibilities regarding the health and safety of the public, we would be justified in restricting competition to those entities that can meet the standards necessary to assure fulfillment of these objectives.

8. Budget

Budget, in simplistic terms, may be broken down into the basic elements of manpower and facilities. As related to manpower, the requirements are necessarily dependent upon the type of quality verification program elected and management's efficiency in implementing such a program. The cost of manpower in a certification system would be equal to that of the basic options or combinations of same (verification of; QA program, QA program implementation, design quality, product quality) plus the additional costs of application processing and certificate issuance control. Facility costs are related to the type of inspection technique utilized. For example, independent testing of motors, valves, etc. would

require significant additional expenditure for the establishment of test facilities or contracted services. Certification by the AEC might also affect facility costs related to the quartering of the additional personnel required to process applications and control certificate issuance.

E. Options Considered in Development of Recommended Quality Verification Programs

The Task Force considered many diverse options during establishment of the recommended AEC quality verification programs to be applied to the various aspects of component fabrication and nuclear power plant construction. These options considered the effective utilization of manpower to accomplish combinations of the various levels of verification, certification programs implemented by other agencies, and the efficacy of the certification concept (as discussed, respectively, in sections B, C, and D of this enclosure). Of foremost importance was the mandate that Regulation must be able to make a finding that the nuclear power plant has been constructed consistent with regulatory requirements and that there is reasonable assurance that the plant can be operated safely. In this regard it is necessary that the selected quality verification program provide due consideration to several key elements; namely:

1. Licensee (be it Utility, Nuclear Steam System Supplier, Architect-Engineer, or Vendor) responsibility for assuring product quality should not be lessened due to the AEC efforts.

2. The AEC quality verification effort should encompass design, fabrication, erection, testing, and operability of components which affect the safety of nuclear power plant operation.
3. The AEC quality verification effort associated with a sphere of activity should be proportional to its importance to safety and incorporate the experience garnered during previous inspection efforts on the part of the AEC and licensee.
4. Detailed specific standards should be used as the barometer against which the findings of the quality verification effort are judged.
5. The AEC should have the necessary freedom of access to all levels of suppliers to ascertain or verify the quality of safety related products used in the nuclear power plant.
6. Enforcement actions should be commensurate with the significance of the violations and, whenever possible, should be directed at the entity which is the underlying cause of the violation.

Several options which may be employed in verifying the quality of the activities performed by vendors, nuclear steam system suppliers, architect-engineers, and at the construction site are presented for comparison.

1. Resident Inspectors

AEC resident inspectors, or combinations of specialized inspectors in residency, can accomplish any quality verification requirements

established by Regulation to the same or greater degree than nonresident inspectors. Utilization of inspectors in this manner, because of the increased in-plant and onsite time, should raise the inspectors' level of knowledge regarding process controls and variables, personnel capabilities, and QA program implementation at a particular shop or construction site. This, in turn, would necessarily increase AEC confidence that the product, including nuclear power plant, was produced consistent with regulatory requirements. There are also some significant drawbacks associated with a resident inspection program. These include program control matters such as maintenance of inspector objectivity and consistency of the inspection effort. Likewise, inspector awareness of generic type issues and keeping abreast of current technological advancements and requirements could be degraded under this option. There is also the concern that resident inspection may downgrade the inspector's chances for promotion. Properly developed and implemented management tools by Regulation, such as audits, inspection guides and training programs, could offset these potential drawbacks.

The efficacy of a resident inspector program is dependent upon the amount and type of nuclear related activity taking place at a certain location and the degree to which the regulatory agency elects to verify the quality of the activity. It is quite obvious

that if only verification of QA program adequacy was elected to be accomplished, resident inspection would not be the route to take. On the other hand, if AEC certification of the quality of each component is the elected approach, resident inspection would essentially be mandatory. Likewise, whereas inspections at a vendor shop having only one nuclear contract should not require resident inspection, it may be desirable to utilize resident inspectors at construction sites or at major vendor shops having numerous nuclear contracts.

2. Third Party Inspection

The utilization of a third party inspection program in meeting our regulatory responsibilities appears to be feasible, if the third party is used in a complementary role. If utilization of a third party inspection program is elected, the AEC, as the agency with the regulatory responsibility, must work closely with the inspecting agency to assure that our regulatory responsibilities are being fulfilled. This may entail AEC audit of the actual inspection effort and assistance in development of inspector training programs, qualification standards, inspection guidance, and documentation requirements. If a third party inspection agency which can more effectively fulfill our regulatory responsibilities is developed, it should be used. On the other hand, a third party inspection agency should not be used just because it exists, or to satisfy

an industry request. In the civilian nuclear industry, the ASME-National Board Inspection system presently verifies the pressure retaining qualities of certain pressure retaining components. There are some deficiencies in this system when compared to Regulatory requirements, but these are being resolved. Once these deficiencies are corrected to AEC satisfaction, it may be desirable to use the ASME-National Board Inspection system to complement the AEC efforts. Likewise, the nuclear industry is attempting to establish a coordinated supplier evaluation program similar to that used in the aerospace field. If this endeavor proves fruitful, we should evaluate and possibly adopt this program as a extension of our own inspection efforts. At the present time, it appears that third party inspection agencies only exist as related to the activities performed in vendor shops and, to a limited extent, erection activities at the construction site. The use of an approved third party could reduce the required AEC inspection effort once confidence in the third party program is established.

3. Specialized Inspectors Based at Regional Offices

The utilization of specialized inspectors operating from one or more specific locations is compatible with any level of quality verification program deemed to be necessary. Whereas resident inspectors would not be the logical choice for inspection of QA

programs, teams of inspectors could effectively accomplish a series of QA inspections at similar facilities, such as valve manufacturing shops or Architect-Engineering offices. On the other hand, if the objective is to verify individual component quality or monitor a large number of varied activities, such as at a nuclear power plant construction site, nonresident inspectors could not be utilized as effectively as resident inspectors.

Advantages of this option include flexibility in utilization of inspectors, availability of more varied technical expertise, ability to respond to unexpected problems, ready identification of generic type deficiencies, and relatively simple programmatic controls. Because of the reduced time spent at the point of inspection, this option does not foster a deep knowledge of the specific process controls, personnel capabilities, QA program implementation, and other variables at a particular shop or construction site.

4. Inspection by Licensee (Utility) or Licensee's Contractors

The AEC presently holds the utility responsible for all aspects of design, component quality, erection, testing, and operation as related to his specific nuclear power plant. This appears to be a good concept, as one party is thereby accountable and "passing-of-the-buck" is minimized. With the increase in the

number of nuclear power plants and the attendant desire to decrease the level of risk associated with any one plant, coupled with the advent of standardization and designated sites, this concept should be modified to some extent. Whenever possible, the licensee, "certificate holder", applicant for designated site, etc., should be held accountable for the activities within his purview. It may be that a Nuclear Steam System Supplier, as the holder of a "certificate" for a standardized plant, purchases certain safety related components. It thereby appears appropriate to require the NSSS to be responsible for the quality of the components so ordered, thus removing the utility from the AEC-required component quality verification process during the purchase/fabrication of those specific components. In the context that the NSSS is a "certificate holder" for a standardized plant, we would not have changed the basic concept of holding the licensee responsible; however, we would have provided more direct AEC access to the point of activity.

Nuclear Power Reactor Incidents and Problems

In order to assess breakdowns in the licensee (utility) quality assurance efforts and weaknesses in Regulation's verification of the utilities' efforts in this regard, the history of incidents and problems experienced at operating reactor and construction sites was examined for a representative period of time. This examination identified weaknesses in the techniques used to meet our regulatory responsibilities and provided one of the bases for the development of the recommended quality verification program to be employed by Regulation.

A. Operating Reactors

Reactor operating licenses require that abnormal occurrences, as defined in the Technical Specifications, be reported to the AEC. Approximately 800 safety related abnormal occurrences were reported to the AEC during the 17 month period used as a sample base (January 1, 1972, to May 31, 1973). Forty percent of the occurrences were traceable in some extent to possible design and/or fabrication related deficiencies. The primary cause of at least 200 of the component malfunctions was design and/or fabrication errors. The remaining incidents were precipitated by operator error, improper maintenance, inadequate erection control, administrative deficiencies, random failure, and variations of the foregoing. Many of the incidents had broad generic applicability and potentially significant consequences. Several of these are discussed in section C of this enclosure.

B. Nuclear Power Plant Construction Sites

Regulation, during inspections of nuclear power plants under construction in the late 1960's and early 1970's, found numerous instances where significant deficiencies in design and construction were detected and corrected by the utility or his contractors. Since many of these deficiencies possessed generic overtones, Regulation determined that certain types of deficiencies detected by the utility prior to issuance of the operating license should be reported to the AEC.

In April 1972, 10 CFR 50.55(e) became effective, thus requiring that the AEC be notified of deficiencies in the design and construction of nuclear power plants, which if uncorrected, could have an adverse effect on the safety of plant operations. Through July 1973, approximately 160 such deficiencies had been reported. Review of approximately one half of the reported deficiencies revealed that the majority of the deficiencies were caused by a breakdown in the licensee or contractor quality assurance program. Examples of such deficiencies are discussed in section C of this enclosure.

C. Incidents Having Generic Implications

The Directorates of Licensing and Regulatory Operations have become aware of many incidents which have generic implications. These incidents were either reported to the Directorates by the licensees, as

required by regulatory requirements, were uncovered during the AEC safety review and inspection programs, or came to the attention of Regulation through some other means, such as an allegation. Once Regulation becomes aware of an incident having generic implications, action is initiated to preclude the occurrence of that type of incident at other nuclear power plants. The following are examples of incidents having generic implications which have come to the attention of Regulation during the past few years.

1. Failed Hangars on ECCS Torus Suction Header for BWR's

   In May 1972, during conduct of the power ascension testing program at a BWR facility, the licensee discovered and reported that several pipe hangars supporting the 24 inch ring suction header for the ECCS systems had failed and the header had sagged approximately six inches. Utility response to the Bulletin issued by Regulatory Operations and followup inspections revealed that similar problems (broken or bent hangar bolts, no lock nuts, improper bolts, overranged seismic restraints, and unbalanced hangar loadings) were in evidence at 4 additional BWR facilities. Failure of the ring suction header could negate operability of the ECCS and constitute a breach of containment integrity. The cause of component failure was attributed to failure to take dynamic effects into consideration during the stress analyses, failure to specify proper bolting materials to

be used in erection of the ring header, and poor workmanship during system erection. Corrective action has been taken at the affected facilities.

2. Limitorque Valve Operators

During 1972, several licensees of light water reactors reported malfunctions in two models of electric valve operators used extensively in safety related systems. The malfunctions were attributed to a lack of proper clearance between the moving parts comprising the torque switch unit and the inability of the torque switch reset spring to return the electrical contacts to a closed position following operation of the valve. During review of the problem, RO found that the vendor had not performed qualification testing to verify the switch design. RO informed all utilities having reactors in operation or under construction of the deficiency. Approximately 70 percent of the facilities had torque switches of the defective model installed. Valve operators utilizing the defective switches (limited to a 2-year manufacturing period) are being equipped with new torque switch assemblies by the licensees and component vendor. The manufacturer has modified his torque switch testing program to preclude repetition of this deficiency which had rendered many safety related valves inoperable.

3. Thin Walled Valves

Inspection of valves in the primary system and engineered safeguards systems at nuclear power plants under construction revealed that valve body castings did not always meet the minimum thickness requirements of the specified codes. This deficiency is attributable to lack of proper quality control at the foundry and failure of the manufacturers to require verification of valve wall thickness. Utilities with reactors in operation are presently being required to verify that valves installed in critical systems have the required wall thickness. Results to date indicate that virtually all facilities are finding valves with wall thickness below code requirements. Current purchase specifications issued by licensees now require positive verification of valve wall thickness.

4. Main Steam Relief System Failures

In a two year period, three incidents associated with main steam system pressure or temperature reduction systems have occurred at PWR facilities. On one occasion the nozzle between the safety valve and the steam line was completely severed during hot functional testing and resulted in injury to seven personnel. During the second incident, 3 of the 4 safety valves had blown off a main steam header and the header was split open during hot functional testing. Eight personnel were injured during the incident. The third incident

involved the decay heat release system. During operation of the decay heat release valve the nozzle backed out of the vent sleeve due to reactive forces. Two personnel died as a result of injuries suffered during the incident. Operator error was a contributing cause of this incident.

The above incidents were precipitated by design inadequacies which did not consider the total dynamic forces involved during valve actuation. As a result of these occurrences, owners of other light water reactor facilities are analyzing and modifying their relief systems as appropriate.

5. Fuel Densification

During inspection of fuel assemblies at an operating PWR facility in April 1972, the licensee observed that an appreciable number of fuel rods had sections with collapsed cladding. This could cause fuel temperatures to exceed expected values both during normal operation and under accident conditions. After initiation of a review of BWR fuel densification effects, other problems associated with fuel densification besides clad collapse were uncovered and resulted in operating limitations being imposed on several operating BWRs. In retrospect, it seems likely that more extensive fuel design evaluation and proof testing, coupled with an increased and continuing post-irradiation examination of fuel rods from older reactors could have revealed the existence of the fuel densification phenomenon.

6. High Energy Line Breaks Outside Containment

An anonymous letter to the ACRS concerning the possible effects of the rupture of the main steam line outside the containment was received in late 1972. This prompted a rather extensive review of all plants regarding the effects of steam line breaks. The reviews initially concentrated on steam lines and compartment pressurization, but quickly expanded to feedwater lines and included pipe whip and the environmental effects of the ruptured pipes. It was determined that the failure of these lines at some plants could have rendered control spaces uninhabitable and safety systems inoperable. Further extension of this review resulted in a rather detailed and lengthy set of preliminary criteria to be applied to present generation plants for all high, moderate, and low energy line breaks outside the containment. A regulatory guide which outlines an acceptable approach as to separation, isolation, and restraint of lines outside the containment whose rupture could cause safety problems is in preparation for the present and future generation plants (expected issuance, early 1974). This design deficiency reinforces the need for Regulation to more thoroughly review the design layout of nuclear power plants.

7. Maximum Drywell Temperature

The maximum design drywell temperature for present generation BWRs was determined from the design basis loss of coolant

accident (LOCA). This is the rupture of the largest recirculation pipe in the primary system and resulted in an equilibrium saturation temperature of about 280°F after blowdown. During a slow release of primary coolant steam at an operating BWR facility, the drywell temperature recorders indicated temperatures in the range of 320 to 340°F. While it was initially suspected that either faulty temperature recorders were the cause of the high temperature or that some highly localized effect was taking place, a simple application of the principles of adiabatic blowdown (constant enthalpy) of the saturated steam associated with a small leak revealed that an equilibrium temperature of 340°F would result. Consequently, the design temperature for the drywell and specified operability temperature for safety related components located in the drywell was changed to 340°F.

8. Flooding of Safety Related Equipment

In June 1972 an expansion joint in the main condenser circulating water system at a BWR facility failed and flooded the turbine building to a depth of approximately 15 feet. The diesel generator cooling water pumps, service water pumps, and the residual heat removal system were flooded and rendered inoperable. Although the failure, per se, was not precipitated by a deficiency in safety related equipment, the inundation of safety related equipment as a result of the failure of a non-safety related component highlighted

a deficiency in over all plant layout. As a result of this occurrence other utilities have examined their plant layouts and corrective actions are being initiated as appropriate. This points up another reason for expanded Regulatory review of plant layout and compartmentalization.

Enclosure 3

Budgetary Implications of Task Force Recommendations

The information presented herein is intended to serve several purposes; first, to establish a professional manpower baseline budget which delineates the budget requirements for case related specific tasks through 1985 assuming that the only manpower changes are due to the growth of nuclear power and that the level of effort per plant remains constant; second, to establish the expanded baseline budget requirements assuming that changes in manpower requirements are precipitated by both the growth of nuclear power and the identified need for the AEC to expand its scope of effort in the areas of design review and inspection; and third, to establish the budget requirements through 1985 assuming the Task Force recommendations contained in this report and in Reference A are adopted, which includes increased near-term manpower increases to establish the site designation and plant standardization concept, and long-term efficiencies to be derived from full implementation of these concepts.

Baseline Budget

A baseline budget was constructed from information gathered from the Regulatory Directorates involved in the licensing and inspection of nuclear power plants. Data were obtained for the professional manpower

presently expended for all of the specific tasks which combine to produce a complete Construction Permit and Operating License review and to perform the AEC quality verification functions associated with the construction and operation of the reactor. No manpower calculations were made for non-CP and OL casework, general Standards efforts, generic work, problem solving, backlog, current work on standardized reviews, and many other Regulatory activities not associated with direct CP and OL review and inspection casework. The data were normalized into man-year work units and grouped according to the four phases of reactor licensing and surveillance; Construction Permit review, Post-CP activities, Operating License review, and Post-OL activities. The basic work units used in the preparation of the baseline budget are presented in Tables I-IV.

The Office of Program Analysis furnished the Task Force with the predicted reactor case workload through 1985, in terms of Construction Permit applications, Operating License applications, and the number of reactors under construction and in operation. This data projection is shown in Table V.

The incremental work units and the projected nuclear growth established the bases for the projected end of year professional staffing levels for each task through 1985. These yearly totals are shown in Tables VI-X as the Baseline data.

The following assumptions and definitions were used in the preparation of the Baseline budget:

1. The Brown Book of August 17, 1973, is assumed to be correct with respect to the number of CP's to be received and issued and the numbers of OL's to be issued in FY 1974.

2. The basic long-range planning target is the most-likely case depicted in WASH 1139(72).

3. Where known, actual capacity of plants is used; where unknown, 1125 MWe per unit is assumed, with an average of 1.8 units per application.

4. On-line electrical capacity through FY 1981 is assumed to be determined by reactors already in-house or projected to be in-house by the end of FY 1974.

5. Any discrepancy between WASH 1139(72) and the forecasted capacity at the end of FY 1981 is reconciled by the end of FY 1985 to be that portrayed by WASH 1139(72). Since more CP applications were received in recent years than would be anticipated to meet the forecasted nuclear power generating capacity in FY 1981, this means that fewer CP applications than might otherwise be anticipated were assumed for FY 1975 through FY 1978 to correct for this anomaly.

6. Wherever the term man-year is used, it has been normalized to represent the total effort expended by one professional person during a year. It includes allowances for such things as annual leave, holidays, other duties, etc., which varies somewhat from one organization to another.

7. The basic staffing levels presented represent those of the professional reviewer and the inspector. They do not include any allowance for needed supervision, management, administrative, or clerical support.

8. The total man-years of effort required for a task is assigned to the year in which the task is begun. For example, the entire effort required for an operating license review is assigned to the year in which the application is received, even though the work may be spread over a two-year span. Also, for operating reactors, OL's issued during the year were all assumed to have been issued on July 1. This was done to simplify the calculations, and the same technique was used for the Baseline, Expanded Baseline, and Recommended calculations, In an equilibrium workload situation the approximation would be perfectly valid; and in an expanding or decreasing workload situation the manpower "difference" determinations will be front-loaded, which will compensate for needed manpower assimilation, thus essentially converting the "man-year" calculation to an "end of fiscal year" staffing level.

Expanded Baseline Budget

Due to the history of problems associated with construction and operation of nuclear plants, the Task Force concluded that the level of participation by the AEC in the area quality verification should be increased, even if the other recommendations of the Task Force regarding site designation and plant standardization are not fully adopted. The Task Force concluded that an additional 2.3 man-years of effort should be devoted to the review of Construction Permit applications, with partiular emphasis placed on qualification testing of components and the design of line routings, plant layout, and equipment location. Similarly, it was concluded that an additional 1.4 man-years should be spent in the evaluation of an Operating License application, with particular emphasis on the design of line routings, plant layout, and operating procedures. Essentially all of the additional manpower recommended for Regulatory Operations is to implement an augmented program of quality verification designed to improve the degree of assurance associated with the operation of nuclear power plants as the number of nuclear power plants increases through the year 2000.

Using these factors, an Expanded Baseline was calculated to represent the manpower requirements necessary to provide the augmentation regarding depth of review and increased quality believed by the Task Force to be warranted.

Recommended Budget

Calculations were also made to determine the budgetary costs associated with full implementation of all of the Task Force's recommendations.

The approach used was to calculate the incremental changes that would occur in the Baseline Budget for each of the several tasks which would be affected by the recommendations. The sum total of all of the incremental changes was algebraically combined with the Baseline Budget values to arrive at the Recommended Budget.

For some activities, such as the custom review of CP applications, the incremental change was negative, i.e., there would be a manpower savings as the custom reviews were phased out. As new activities were phased in, the increments were positive, indicating a need for additional manpower to perform these revised functions.

The work units used for the new activities were formulated from the basic work units furnished for the Baseline Budget, with some modifications to allow for the efficiencies deriving from the new activities. For example, it is estimated that the effort to review a site and carry it through the designation process as a single operation will be only 75% of the effort required to take the same review through a two step process as is presently the case.

In developing the work load assumptions for the Recommended Budget, the reactor case load was assumed to be the same as the baseline. As certain ways of doing business were phased out and new procedures phased in, the case load was reallocated accordingly. For example, custom reviews of

CP's will be almost phased out by FY 78 and almost all site evaluations will be done as Designated Sites beginning in FY 77.

Summary Results

Tables VI-X depict the projected Baseline Budget, the Expanded Baseline Budget where appropriate, and the Recommended Budget. Table VI summarizes the budgetary data for the combined activities of Technical Review, Reactor Projects, Regulatory Operations, and Regulatory Standards that would be affected by the Task Force recommendations. As previously noted, the values shown represent the number of man-years of professional effort applied to each task, which are directly convertable to EOFY augmented staffing needs, because of the manpower front loading that was assumed. Any extrapolation to total Regulatory manpower requirements (such as the FY 1974 and FY 1975 budgets) must recognize that the data do not include supervision and management, clerical support, administrative services, backlog, or any of the ancillary activities which are peripheral to the principal work of application review and inspection.

Tables VII through X depict the summaries of budgetary impact on each of the four organizations affected by the recommendations - Technical Review (Table VII), Reactor Projects (Table VIII), Regulatory Operations (Table IX), and Regulatory Standards (Table X).

Table I

BASIC WORK UNITS

Construction Permit Review

Review Organization	Activity	Work Unit
TR	Environmental Review	0.65 man-years/appl
TR	Environmental Hearings	.10
TR	Site Safety Review	.85
TR	Site Safety Hearings	.10
TR	Subtotal	1.70
RP	Environmental Review	.59
RP	Environmental Hearings	.13
RP	Site Safety Review	.30
RP	Safety Hearings	.10
RP	Subtotal	1.12
	TOTAL SITE REVIEW	2.82
TR	Nuclear Island Review	2.12
TR	Balance of Plant Review	.36
TR	Safety Hearings	.10
TR	Subtotal	2.58
RP	Safety Review	.51
RP	Safety Hearings	.21
RP	Subtotal	0.72
	TOTAL PLANT REVIEW	3.30
RO	Preconstruction Inspection	0.13
	TOTAL CP REVIEW	6.25

Table II

BASIC WORK UNITS

Post-CP Issuance Effort

Review Organization	Activity	Work Unit
	(Reactors in first year of construction)	
RP	Environmental Review	0.11 man-years/reactor
RP	Safety Review	.32
RP	Subtotal	.43
	(Reactors in first 4 years of construction)	
RO	Early Construction Inspection	0.59 man-years/reactor/year
	(Reactors in fifth year of construction)	
RO	Construction Inspection and Preoperational Testing	2.0 man-years/reactor
RP	Preoperational Review	0.22
	(Entire Program)	
RO	Vendor, NSSS, A/E Inspection	10.0 man-years/year

Table III

BASIC WORK UNITS

Operating License Review

Review Organization	Activity	Work Unit
TR	Environmental Review	0.52 man-years/appl
TR	Environmental Hearings	.07
TR	Site Safety Review	0.53
TR	Site Safety Hearings	.10
TR	Subtotal	1.22
RP	Environmental Review	.59
RP	Environmental Hearings	.07
RP	Site Safety Review	.35
RP	Safety Hearings	.10
RP	Subtotal	1.11
	TOTAL SITE REVIEW	2.33
TR	Nuclear Island Review	2.66
TR	Balance of Plant Review	.38
TR	Safety Hearings	.10
TR	Subtotal	3.14
RP	Safety Review	.68
RP	Safety Hearings	.21
RP	Subtotal	.89
	TOTAL PLANT REVIEW	4.03
	TOTAL OL REVIEW	6.36

Table IV

BASIC WORK UNITS

Post-OL Effort

Review Organization	Activity	Work Unit
	(Reactors in first year of operation)	
RP	Operating Reactors (assuming half are additional units)	0.33 man-years/reactor
RO	Initial Operation Inspection	1.7
	(Reactors in second and third years of operation)	
RP	Operating Reactors	.33 man-years/reactor/year
RO	Routine Operation Inspection	1.1
	(Reactors after third year of operation)	
RP	Operating Reactors	.20
RO	Routine Operation Inspection	1.1
	(All reactors)	
TR	Operating Reactors	0.28
RP	Environmental Review	0.13

TABLE V

PROJECTED POWER REACTOR WORKLOAD FOR BASELINE PLANNING

	FY 74	FY 75	FY 76	FY 77	FY 78	FY 79	FY 80	FY 81	FY 82	FY 83	FY 84	FY 85
CP Applications	16(32)	14(26)	10(17)	12(21)	14(24)	19(34)	21(37)	22(40)	23(42)	25(44)	26(47)	28(49)
OL Applications	6(8)	6(8)	3(4)	13(23)	14(30)	16(32)	14(26)	10(17)	12(21)	14(24)	19(34)	21(37)
Units First 4 Years Const	55	69	93	111	105	96	88	96	116	135	153	163
Units Last Year Const	23	16	8	8	4	23	30	32	26	17	21	24
Units First Year Operation	12	23	16	8	8	4	23	30	32	26	17	21
Units 2nd & 3rd Yr Opn	6	13	35	39	24	16	12	27	53	62	58	43
Units After 3rd Yr Opn	13	18	19	31	54	70	78	86	90	113	143	175
Total Operational Units (EOY)	54	70	78	86	90	113	143	175	201	218	239	263

TABLE VI
IMPACT OF RECOMMENDATIONS, REGULATION SUMMARY

	FY 74	FY 75	FY 76	FY 77	FY 78	FY 79	FY 80	FY 81	FY 82	FY 83	FY 84	FY 85
BASELINE												
Technical Review	103.32	101.20	75.48	129.83	145.04	176.28	182.56	177.80	199.76	224.32	255.16	278.32
Reactor Projects	68.96	76.68	69.65	92.87	94.49	113.35	125.40	136.74	156.75	172.71	190.21	201.91
Regulatory Operations	131.83	157.73	168.77	183.65	181.17	216.51	262.75	308.80	345.13	363.60	395.65	433.31
Regulatory Standards	41.00	64.00	97.00	99.00	103.00	100.00	95.00	90.00	84.00	66.00	45.00	45.00
Total	345.09	399.61	410.90	505.40	523.70	606.1[illegible]	665.71	713.34	785.64	826.63	886.02	961.54
INCREMENTS RECOMMENDED												
Technical Review	98.04	120.22	120.04	154.06	139.56	141.87	112.50	103.28	88.24	69.14	55.05	7.87
Reactor Projects	10.00	21.10	25.10	38.38	25.10	26.76	15.79	11.89	2.19	(7.96)	(17.55)	(38.59)
Regulatory Operations	123.87	171.67	233.43	254.07	236.36	252.39	219.60	208.82	193.21	176.78	174.17	160.54
Regulatory Standards	2.00	18.00	27.00	32.00	29.00	25.00	17.00	9.00	4.00	0.00	0.00	0.00
Total	233.91	330.99	405.57	478.51	430.02	446.02	364.89	332.99	287.64	237.96	211.67	129.82
NEW TOTAL RECOMMENDED	579.00	730.60	816.47	983.91	953.72	1052.16	1030.60	1046.33	1073.28	1064.59	1097.69	1091.36
EXPANDED BASELINE INCREMENTS												
Technical Review	45.20	40.60	27.20	45.80	51.80	66.10	67.90	64.60	69.70	77.10	86.40	93.80
Regulatory Operations	123.87	131.67	228.43	249.07	231.36	250.57	224.38	221.46	215.63	243.17	259.21	261.86
Total	169.07	172.27	255.63	294.87	283.16	316.67	292.28	286.06	285.33	320.27	345.61	355.66
EXPANDED BASELINE	514.16	571.88	666.53	800.27	806.86	922.81	957.99	999.40	1070.97	1146.90	1231.63	1317.20

TABLE VII
IMPACT OF RECOMMENDATIONS, TECHNICAL REVIEW
SUMMARY

	FY 74	FY 75	FY 76	FY 77	FY 78	FY 79	FY 80	FY 81	FY 82	FY 83	FY 84	FY 85
BASELINE												
TR CP Review	68.48	59.92	42.80	51.36	59.92	81.32	89.88	94.16	98.44	107.00	111.28	119.84
TR OL Review	26.16	26.16	13.08	56.68	61.04	69.76	61.04	43.60	52.32	61.04	82.84	91.56
TR Operating Reactors	8.68	15.12	19.60	21.84	24.08	25.20	31.64	40.04	49.00	56.28	61.04	66.92
Total	103.32	101.20	75.48	129.88	145.04	176.28	182.56	177.80	199.76	224.32	255.16	278.32
INCREMENTS RECOMMENDED												
CP Reviews	36.16	(1.06)	(10.10)	(31.74)	(59.92)	(81.32)	(89.88)	(94.16)	(98.44)	(107.00)	(111.28)	(119.84)
PBCP Reviews	0	10.90	10.90	19.62	30.52	41.42	10.90	10.90	10.90	10.90	10.90	10.90
OL Reviews	8.88	8.88	4.44	19.24	20.72	23.68	(8.48)	(14.40)	(34.80)	(61.04)	(82.84)	(91.56)
PBOL Reviews	0	0	0	0	0	0	11.85	11.85	21.33	33.18	45.03	11.85
SNI Review	53.00	79.60	92.90	106.20	119.50	119.50	119.50	119.50	119.50	119.50	119.50	119.50
Designated Site Review	0	21.90	21.90	41.61	30.66	41.61	45.99	48.18	50.37	54.75	56.94	61.32
Ref. Design Approval	0	0	0	0	0	0	27.68	29.41	31.14	34.60	36.33	39.79
Operating Reactors	0	0	0	(0.87)	(1.92)	(3.02)	(5.06)	(8.00)	(11.76)	(15.75)	(19.53)	(24.09)
Total	98.04	120.22	120.04	154.06	139.56	141.87	112.50	103.28	88.24	69.14	55.05	7.87
NEW TOTAL	201.36	221.42	195.52	283.94	284.60	318.15	295.06	281.08	288.00	293.46	280.21	286.19
EXPANDED BASELINE INCREMENTS												
Add for each CP 2.3 MY	36.80	32.20	23.00	27.60	32.20	43.70	48.30	50.60	52.90	57.50	59.80	64.40
Add for each OL 1.4 MY	8.40	8.40	4.20	18.20	19.60	22.40	19.60	14.00	16.80	19.60	26.60	29.40
EXPANDED BASELINE	148.52	141.80	102.68	175.68	196.84	242.38	250.46	242.40	269.46	301.42	341.56	372.12

TABLE VIII
IMPACT OF RECOMMENDATIONS, REACTOR PROJECTS
SUMMARY

	FY 74	FY 75	FY 76	FY 77	FY 78	FY 79	FY 80	FY 81	FY 82	FY 83	FY 84	FY 85
BASELINE												
RP CP Review	29.44	25.76	18.40	22.08	25.76	34.96	38.64	40.48	42.32	46.00	47.84	51.52
RP Post-CP Effort	14.95	16.42	15.52	12.94	8.19	14.09	16.92	21.66	21.63	20.94	22.68	24.20
RP OL Effort	12.00	12.00	6.00	26.00	28.00	32.00	28.00	20.00	24.00	28.00	38.00	42.00
RP Post-OL Effort	12.57	22.50	29.73	31.85	32.54	32.30	41.84	54.60	68.80	77.77	81.69	87.19
Total	68.96	76.68	69.65	92.87	94.49	113.35	125.40	136.74	156.75	172.71	190.21	204.91
INCREMENTS RECOMMENDED												
RP CP Review	0	(9.20)	(9.20)	(16.56)	(25.76)	(34.96)	(38.64)	(40.48)	(42.32)	(46.00)	(47.84)	(51.52)
RP PBCP Review	0	3.60	3.60	6.48	10.08	13.68	3.60	3.60	3.60	3.60	3.60	3.60
RP OL Review	0	0	0	0	0	0	(10.00)	(10.00)	(18.00)	(28.00)	(38.00)	(42.00)
RP PBOL Review	0	0	0	0	0	0	4.45	4.45	8.01	12.46	16.91	4.45
RP SNI Review	10.00	10.00	14.00	18.00	20.00	20.00	20.00	20.00	20.00	20.00	20.00	20.00
RP DS Review	0	16.70	16.70	31.73	23.38	31.73	35.07	36.74	38.41	41.75	43.42	46.76
RP Ref. Design Review	0	0	0	0	0	0	8.00	8.50	9.00	10.00	10.50	11.50
RP Operating Reactors	0	0	0	(1.27)	(2.60)	(3.87)	(6.69)	(10.92)	(16.51)	(21.77)	(26.14)	(31.38)
Total	10.00	21.10	25.10	38.38	25.10	26.76	15.79	11.89	2.19	(7.96)	(17.55)	(38.59)
NEW TOTAL	78.96	97.96	94.75	131.25	119.59	140.11	141.19	148.63	158.94	164.75	172.66	166.32

TABLE IX
IMPACT OF RECOMMENDATIONS, REGULATORY OPERATIONS
SUMMARY

	FY 74	FY 75	FY 76	FY 77	FY 78	FY 79	FY 80	FY 81	FY 82	FY 83	FY 84	FY 85
BASELINE												
Preconstruction Insp.	2.08	1.82	1.30	1.56	1.82	2.47	2.73	2.86	2.99	3.25	3.38	3.64
Early Construction Insp.	32.45	40.71	54.87	65.49	61.95	56.64	51.92	56.64	68.44	79.65	90.27	96.17
Preop. Test Insp.	46.00	32.00	16.00	16.00	8.00	46.00	60.00	64.00	52.00	34.00	42.00	48.00
Vendor/NSSS/AE Insp.	10.00	10.00	10.00	10.00	10.00	10.00	10.00	10.00	10.00	10.00	10.00	10.00
Operating Reactors	41.30	73.20	86.60	90.60	99.40	101.40	138.10	175.30	211.70	236.70	250.00	275.50
Total	131.83	157.73	168.77	183.65	181.17	216.51	262.75	308.80	345.13	363.60	395.65	433.31
INCREMENTS RECOMMENDED												
Additions												
Early Construction Insp.	58.85	73.83	99.51	118.77	112.35	102.72	80.08	87.36	105.56	122.85	139.23	148.33
NSSS/AE Inspection	0.00	8.00	18.00	18.00	18.00	18.00	18.00	18.00	18.00	18.00	18.00	18.00
Mfg. Inspection	0.00	32.00	72.00	72.00	72.00	72.00	72.00	72.00	72.00	72.00	72.00	72.00
Preop. Test Insp.	40.02	27.84	13.92	13.92	6.96	40.02	41.40	44.16	35.88	23.46	29.98	33.12
Operating Reactors	20.00	20.00	20.00	20.00	20.00	20.00	20.00	20.00	20.00	20.00	20.00	20.00
Reliability Analysis	5.00	10.00	10.00	15.00	15.00	15.00	20.00	20.00	20.00	25.00	25.00	25.00
Subtractions												
NSSS/AE Insp (5%/yr efficiency)	0.00	0.00	0.00	0.00	0.00	0.00	(0.90)	(1.80)	(2.70)	(3.60)	(4.50)	(4.50)
Mfg. Insp. (5%/yr efficiency)	0.00	0.00	0.00	0.00	0.00	0.00	(3.60)	(7.20)	(10.80)	(14.40)	(18.00)	(18.00)
Operating Reactors (4%/yr efficiency)	0.00	0.00	0.00	(3.62)	(7.95)	(12.17)	(22.10)	(35.06)	(50.81)	(66.28)	(80.00)	(99.18)
Early Construction (2%/yr efficiency)	0.00	0.00	0.00	0.00	0.00	(3.18)	(5.28)	(8.64)	(13.92)	(20.25)	(27.51)	(34.23)
Total	123.87	171.67	233.43	254.07	236.36	252.39	219.60	208.82	193.21	176.78	174.17	160.54
NEW TOTAL	255.70	329.40	402.20	437.72	417.53	468.90	482.35	517.62	538.34	540.38	569.82	593.85
EXPANDED BASELINE INCREMENTS	123.87	131.67	228.43	249.07	231.36	250.57	224.38	221.46	215.63	243.17	259.21	261.86
EXPANDED BASELINE	255.70	289.40	397.20	432.72	412.53	467.08	487.13	530.26	560.76	606.77	654.86	695.17

TABLE X
IMPACT OF RECOMMENDATIONS, REGULATORY STANDARDS
SUMMARY

	FY 74	FY 75	FY 76	FY 77	FY 78	FY 79	FY 80	FY 81	FY 82	FY 83	FY 84	FY 85
BASELINE												
LWR	38.00	59.00	85.00	82.00	80.00	70.00	51.00	33.00	25.00	25.00	24.00	24.00
HTGR	3.00	5.00	12.00	17.00	23.00	30.00	44.00	57.00	59.00	41.00	21.00	21.00
Total	41.00	64.00	97.00	99.00	103.00	100.00	95.00	90.00	84.00	66.00	45.00	45.00
INCREMENTS RECOMMENDED												
Qualification Test Stds.	0.00	3.00	5.00	5.00	4.00	4.00	0.00	0.00	0.00	0.00	0.00	0.00
Qualification Test Facilities	1.00	2.00	2.00	1.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Siting Guides and Regs	1.00	3.00	3.00	2.00	2.00	2.00	2.00	1.00	1.00	0.00	0.00	0.00
Regional Siting Guides & Regs	0.00	6.00	6.00	6.00	6.00	6.00	6.00	5.00	3.00	0.00	0.00	0.00
Stds. for Comprehensive License Review	0.00	2.00	5.00	8.00	6.00	4.00	2.00	0.00	0.00	0.00	0.00	0.00
QA Stds. for Fabrication and Construction	0.00	2.00	5.00	8.00	6.00	4.00	2.00	1.00	0.00	0.00	0.00	0.00
Preop. Test Stds. for Standardized Plant	0.00	0.00	1.00	2.00	5.00	5.00	5.00	2.00	0.00	0.00	0.00	0.00
Total	2.00	18.00	27.00	32.00	29.00	25.00	17.00	9.00	4.00	0.00	0.00	0.00
NEW TOTAL	43.00	82.00	124.00	131.00	132.00	125.00	112.00	99.00	88.00	66.00	45.00	45.00