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ANALYSIS BY SIMULATION OF THE DISPOSITION OF NUCLEAR FUEL WASTE

by

Jeffery Lee Turek

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All artwork was performed by .

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CHAPTER ONE - INTRODUCTION

If power shortages are to be avoided, spent nuclear fuel storage facilities must be provided during the early 1980's. Due to the social, economical, technical, and political issues surrounding nuclear power some form of a managerial aid is required to enhance the decision-making process. A simulation model of the back-end of the nuclear fuel cycle is developed. This model assists the Department of Energy managers in the complex decision processes embroiled with solving spent fuel storage demands.

During either the hottest summer days or coldest winter nights, utilities in certain regions of the country are producing electricity at near-maximum capability. Hence, the untimely loss of any electricity generating facility can cause severe consequences. Although nuclear power accounts for only approximately eleven percent of the electrical needs of this country, a reduction in nuclear energy can result in power shortages and brownouts. Presently, the future of the nuclear industry rests in solving the political and technological problems associated with the back-end of the nuclear fuel cycle.

These problems are compounded by the President's energy announcement of April, 1977. This announcement states that, in order to meet the non-proliferation objectives of the United States, reprocessing of spent nuclear fuel is discontinued. In addition, a limited amount of foreign spent nuclear fuel is to be stored in the continental United States. Since practically all existing at-reactor storage facilities were designed with the anticipation of reprocessing, this shift in the policy of the United States mandates that some form of interim spent fuel storage be provided until the acquisition of final storage repositories is realized (first repository scheduled to be on-line in 1997). There exists an urgent; nay, a dire need to provide interim storage space in order that no commercial nuclear reactor facility has to shutdown due to its inability to store spent nuclear fuel.

This country does not have the time nor the luxury to apply inefficient or trial and error management techniques to the myriad of factors related to the nuclear fuel cycle. To enlist the most efficient method with which to aid both the Department of Energy managers and the utility managers in the decision-making processes, a simulation model is developed which permits examination of all the factors and permutations thereof which affect the back-end of the nuclear fuel cycle. A sampling of such factors includes: 1) on-line times and handling rates of the Away-From-Reactor (AFR) Storage facilities, 2) spent fuel output of individual reactors, 3) spent fuel storage demands from foreign countries, and 4) repository on-line times and capacities. Each factor is a culmination of literally hundreds of activities, each of which are trackable via the simulation. A simplified example of a related ladder of activities is: Public hearings and comments are required to complete a generic environmental impact statement covering interim storage facilities. After the generic statement is issued and a Record of Decision is filed, a site-specific environmental impact statement is prepared followed by facility design, then facility construction, and finally operation. One managerial question might be "How does a delay of six months in public hearings affect the demand for repository space in the year 2005?" The simulation model quickly and inexpensively answers such a question. The model permits the scrutiny of any desired scenario, arming the managers with a most efficient and effective managerial technique--absolutely essential if this country is going to solve the current nuclear dilemma in a timely fashion.

Initial results of the model using input data based upon the most probable energy scenario as conceived by Department of Energy officials indicate that certain reactor complexes require away-from-reactor storage prior to the availability of any federal interim storage facility. To prevent the actual shutdown of these reactors, temporary measures are being taken. These stopgap measures include the storage of spent fuel against the full core reserve capacity of at-reactor storage pools and the transshipment of spent fuel to other reactors possessing available storage space.

This effort details the steps involved in the development of the spent fuel simulation model. Since the work combines the techniques associated with both nuclear engineering and operations research, a section on the current nuclear situation and a section on simulation are included in Chapter Two. The second chapter also examines the availability of the different sources of input data. Chapter Three covers the simulation model development and verification. The author believes that since this model has been designed for implementation by the Department of Energy, that accurate and accountable input is essential. Therefore, Chapter Four discusses the various input sources in detail and investigates the techniques developed to ensure the accuracy and accountability of said data. Chapter Five contains results and Chapter Six examines the conclusions. Finally, Chapter Seven states recommendations for future research.

CHAPTER TWO - BACKGROUND

A simulation model is developed which aids Department of Energy (DOE) managers, Dupont managers, and utility managers in the realization of the DOE Away-From-Reactor (AFR) Spent Fuel Storage Program objective. The objective states that no commercial power reactor will have to shut down due to a lack of spent fuel storage space.[1] This model simulates the back-end of the nuclear fuel cycle as defined by the United States Policy of 1977. In the development of the simulation model numerous subject matters are examined, the data digested, and the pertinent information extracted. This chapter reviews each of the three major areas of research which affect the model. The area, THE CRITICAL NEED, explains the urgency which exists in solving the storage problem. Next, THE APPROACH, outlines the principles underlying simulation techniques and why simulation provides a most powerful tool in the analysis of the fuel cycle. In addition, the advantages and disadvantages of various simulation languages are reviewed. Once a need for a model and a modelling technique are chosen, a source of accurate data must be obtained. The third area, THE INPUT, examines the sources of input data. Possessing all the proper prerequisites, the actual development of the simulation model is covered in Chapter Three.

THE CRITICAL NEED

One thought repeatedly surfaces when reviewing literature concerning the status of spent fuel storage. This thought concerns just how did this country get into such a dire situation. Throughout the history of nuclear power, fuel discharge projections have been made. What happened? To understand the current dilemma, one must first understand that nuclear power is a vital energy source which cannot be abolished. Next, an understanding of the difference between the nuclear fuel cycle as conceived prior to 1977 and the currently defined nuclear fuel cycle is required. Finally, an examination of existing and planned storage facilities, their availability, and methods to increase the capacities of the spent fuel storage pools is given. This section includes both at-reactor (AR) storage facilities and away-from-reactor (AFR) storage facilities. At this point, the mystery should begin to unravel.

Nuclear Role:

In 1979 the total energy consumption of the United States was 79 quads (one quad is one quadrillion BTU).[2] Of this amount, 24 quads (approximately thirty percent) was produced by the electric utilities.[2] A further breakdown of the production of energy by the electric utilities indicates that nuclear facilities produced 2.7 quads (eleven percent); hydroelectric facilities, 3.1 quads (thirteen percent); petroleum facilities, 3.6 quads (fifteen percent); natural gas facilities, 3.6 quads (fifteen percent); and coal facilities, 11.3 quads (forty-six percent).[2] A breakdown by the end users of electricty shows thirty-four percent for residential, forty-one percent for industrial, and twenty-five percent for commercial users.[3] A regional breakdown reveals that the nuclear contribution was greatest in the Northeast, Mid-West, and Mid-Atlantic regions.[3] An examination of these figures attest to the vital role that nuclear power plays in the overall energy picture of the United States. A loss or severe reduction in the nuclear generation capabilities of this country can affect all sectors of our economy.

Several issues highlight the plight of nuclear power.

One: during peak loads there is not enough non-nuclear capacity to make-up for a serious loss of nuclear power. Two: an energy system comprising a balance of different energy sources has, historically, proven to be most advantageous. Two Midwest states, Illinois and Indiana, had drastically different effects during the coal strike of 1977 and 1978. The electrical energy for Indiana is produced predominately from coal; whereas, a balance of coal and nuclear supplies Illinois. During the strike, Indiana had extensive plant and school closings, whereas Illinois saw only minor disturbances. Three: coal, which some individuals proclaim should replace nuclear power, produces hazardous environmental effects. Acid rain and the "greenhouse" effect are potentially more damaging than any accident with nuclear energy. Four: producing oil from oil shale requires vast amounts of water, water which is just not going to be available.[3] Hence, nuclear energy is a vital resource that this country can ill-afford to do without.

Nuclear Fuel Cycle:

The critical need for storage space can be observed by studying the nuclear fuel cycle, as conceived by the utilities prior to the change in the United States's energy policy of 1977. Prior to 1977, reprocessing of spent fuel was anticipated. Since reprocessing returns fuel to the reactor the term "cycle" is employed. Figure 1 depicts the nuclear fuel cycle as conceived prior to 1977.[4] Each stage of this cycle is briefly examined starting with mining and milling.

Almost all uranium ore mined in the United States comes from west of the Mississippi River, notably the Colorado Plateau and Wyoming basin areas. The uranium content of these ores typically ranges from 0.15 to 0.30 percent uranium oxide. The uranium isotope concentrations in the oxide are 99.3 percent U-238 and 0.7 percent U-235. After mining, the uranium ore is milled, purified, and sent to the conversion plants for refinement. Here, the ore is converted into a gaseous compound termed uranium hexaflouride. This compound is required for the next phase of the cycle, enrichment. Enrichment is the process where the percentage of U-235 is increased above its natural value.[5] Enrichment is needed since commercial reactors require approximately three percent fissionable U-235. Presently, commercial fuel is enriched via gaseous diffusion. Small scale experiments have been conducted

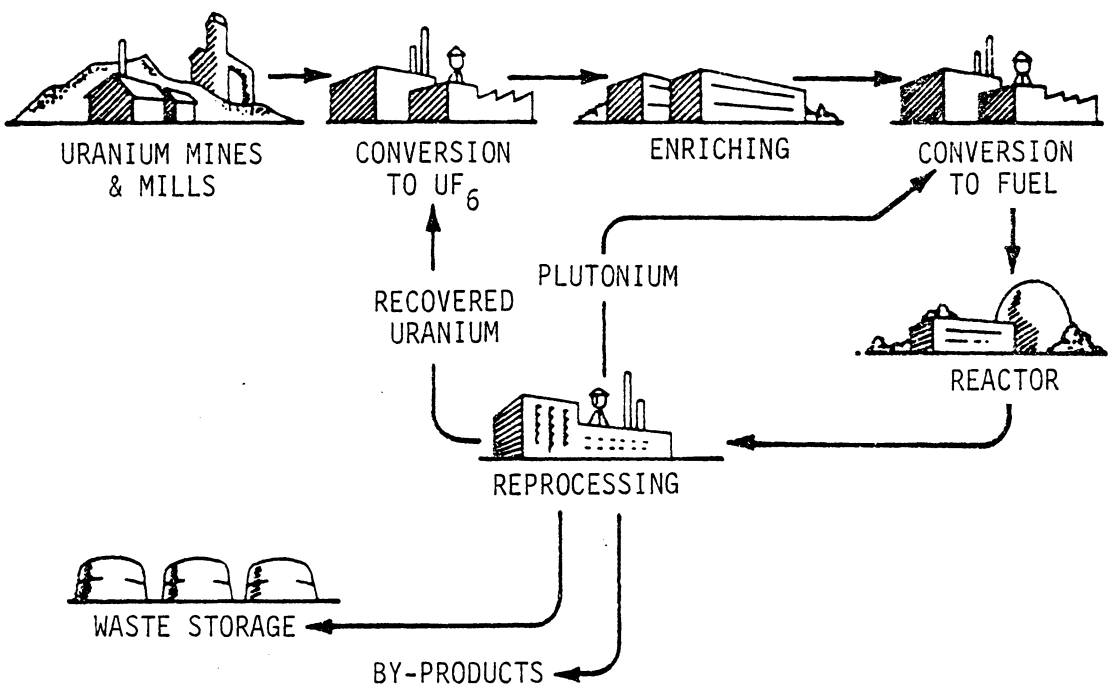


Figure 1. Nuclear Fuel Cycle, pre-1977

enriching uranium by gaseous centrifuge techniques. Since these experiments have proven satisfactory, a commercial size centrifugal plant is being constructed in Portsmouth, Ohio. The centrifuge process holds promise to lower enrichment costs; largely due to the tremendous reduction in electricity required for operation.

After enrichment, the conversion and fabrication of fuel for use in a reactor core are accomplished. First, the uranium hexaflouride is converted into the fuel material; typically, uranium dioxide. Next, the uranium dioxide is fabricated into small fuel pellets which are loaded into individual fuel rods. In turn, the fuel rods are grouped into fuel assemblies. Table 1 outlines the characteristics of a typical fuel assembly.[6] A commercial reactor can have upwards to several hundred assemblies comprising the core.

One-third of the core of a commercial reactor is replaced each year with fresh fuel assemblies. Hence, an individual fuel assembly has a life, within the reactor core, of three years. Upon the removal of a fuel assembly (now termed spent fuel assembly), the assembly is stored at a storage pool located adjacent to the reactor containment building. Such a storage facility is termed an at-reactor (AR) storage pool. Presently, all spent

Table 1. Characteristics of Typical LWR Fuel Assemblies

	PWR	BWR
Assembly length, m	4.1	4.4
Assembly width, cm	21.3	14.0
No. of fuel rods	264 (17x17 array)[a]	64 (8x8 array)
Fuel rod diameter, cm	.94	1.24
Uranium weight, Metric ton	0.52	0.21
Total assembly weight, Metric ton	0.66	0.28

[a] Array -- the regular arrangement.

fuel is stored as fuel assemblies; although, some experimentation is underway to store fuel by individual rods.[6] Table 2 lists some characteristics of AR storage pools.[6] The AR storage facilities are sized to hold an amount of assemblies equal to the number of assemblies comprising one full core plus one reload (one-third core). This size is realized since fuel is to be shipped to chemical reprocessing plants. A section of storage capacity in each pool remains available in case of an emergency requiring the removal of the entire reactor core. This space is termed full core reserve (FCR) and, although not required by federal or state regulation, is highly desirable. Reactors have had to make full core discharges fifty times in the past.[6] Some studies suggest that consumer rates may be higher if a utility does not possess FCR space. When a full core discharge is required and there is not sufficient on-site storage, then permission must be obtained to transship the fuel. The time required for approval translates into time which replacement power must be purchased from another utility.[6]

Spent fuel assemblies are stored at the reactor for a period of time to allow a reduction in the radioactivity levels. Although the minimum time for storage is 120 days, longer cooling periods reduce the radiolysis of some

Table 2. Characteristics of At-Reactor Spent-Fuel Storage Pools

SIZE	as small as 3.3m by 3.3m (LaCross Reactor) as large as 12.2 m by 18.3 m (Donald C. Cook Reactor)
DEPTH	as shallow as 7.9m (Humbolt Bay Reactor) as deep as 13.7m (Donald C. Cook Reactor) at least 3.7m of water above fuel
WATER CHEMISTRY	maintained below 50°C less than $5X10^4$ curies per cubic meter
MATERIALS	reinforced concrete lined with stainless steel

of the chemical reagents used in reprocessing. Hence, the reprocessing of older fuel (stored for periods up to five years or more) is desirable. After this storage, the spent fuel is processed into uranium, plutonium, and waste fractions. Plutonium comes from nuclear activation of U-238, forming Pu-239. Pu-239, in itself, is an excellent fissionable material; in fact, since Pu-239 is the primary constituent in nuclear weapons, the misuse of this element is the basis for the halt in reprocessing. The cycle is completed when the recovered uranium is converted to uranium hexafluoride for enrichment purposes and then combined with plutonium into new fuel rods.

Spent Fuel Storage:

In summary, the preceeding description of a nuclear fuel cycle applies to a cycle as was envisioned prior to the United States Administration's energy policy of 1977. Uranium, plutonium, and their isotopes make up approximately seventy percent of the weight of a spent fuel assembly. The remaining thirty percent consists of structural materials and fission products. Of the uranium and plutonium, typically one percent is U-235 and one percent is Pu-239. After a period of storage, this fuel was originally expected to be reprocessed and cycled through the reactor. With a halt in reprocessing, this spent fuel must be stored, never again to see a reactor core. The present ban on reprocessing mandates the need for large capacities of spent fuel storage space. The present objective of the AFR Program is to guarantee that no commercial reactor has to shut down due to a lack of spent-fuel storage space.[1] Geologic repositories which are being designed for both defense wastes and for commercial spent fuel will not be on-line until 1997 at the earliest.[7] Therefore, since AR storage pools are designed for reprocessing and do not possess the capacity for long-term storage, some form of interim storage must be forthcoming. To meet the stated objective of the AFR program, these AFR facilities must start coming on-line in the early 1980's. This date is chosen since several reactors are projected to fill their available AR pools in the early 1980's.[7]

The AFR Program objectives are being pursued via a variety of fronts: 1) increase AR storage by new storage techniques; 2) investigate benefits of intra- and inter-utility shipment of spent fuel; 3) license and modify, if necessary, existing nuclear facilities which are potentially capable of storing spent fuel until storage repositories come on-line; and 4) design, construct, and operate new AFR facilities as needed. Each area of endeavor is explained in the following paragraphs.

AR Storage Facilities:

Typically, AR storage racks are designed using aluminum or steel and result in a fuel storage density of 2.69 metric tons of uranium per square meter (mtu/sq m). Such a density prevents criticality, even if unirradiated U-235 is stored in the pools. In order to increase the storage density, a neutron absorbing substance (poison) can be added to the racking material. Presently, three types of new design racks are licensed. The first design employs stainless steel and achieves a fuel storage density of 4.20 mtu/sq m. An increase in storage density to 6.24 mtu/sq m is achieved if boron (a poison) is added. Finally, boral racks comprised of boron carbide dispersed in aluminum achieve the same storage density of 6.24 mtu/sq m.[6]

Studies are underway to increase further the storage density. One method consists of placing individual fuel assemblies in cans with small metal beads surrounding the fuel rods. This method permits a density of 8.07 mtu/sq m. Another technique involves the uniform crushing of the fuel spacers and grid plates within a fuel assembly until the rods are almost touching. Thus, a storage density of 10.23 mtu/sq m can be achieved. The most promising new development involves removing the individual fuel rods from an assembly and placing them in a can. Within this can the fuel rods would be touching each other, achieving a density of 11.84 mtu/sq m.[6] None of these newer methodologies have been licensed.

Transshipment of Fuel:

This method of allocation of fuel storage space is almost entirely political rather than technological. If an individual reactor's AR pool is full, then an option exists to ship its spent fuel to either a storage pool with available capacity located within the same utility (intrashipment) or a storage pool owned by another utility (intershipment). The intershipment option is considered only as a last resort.

Each shipment must meet the approval of the Nuclear

Regulatory Commision (NRC). In addition, states and localities may require approval. Legal proceedings brought on by environmentalists or any other group could hinder transshipment. The present complexities of this method of storage enhancement reduce the effectiveness of transshipment as an efficient, long-term storage aid.

AFR Facilities:

The President's Message to Congress on Feburary 12, 1980 directed the Department of Energy (DOE) to design, acquire or construct, and operate one or more away-from-reactor facilities and to accept, for storage, domestic spent fuel and a limited amount of foreign spent fuel until permanent disposal facilities are available.[8] The DOE is considering the acquisition of three existing facilities, originally intended as fuel reprocessing facilities, and the construction of one or more new AFR facilities. The three existing facilities are: 1) Barnwell Nuclear Fuel Plant (BNFP); 2) General Electric-Morris Plant (GE-Morris); and 3) Western New York Nuclear Service Center, WNYNSC (West Valley). Each facility is described below.

The Allied General Nuclear Services (AGNS) Fuel Recovery and Storage Station is part of the Barnwell Nuclear Fuel Plant Separation Facility, located in Barnwell, South Carolina. The BNFP is designed to reprocess spent fuel and provide for the proper management of the wastes resulting from such reprocessing. This facility has never been licensed for reprocessing. In 1974 a license application was submitted to employ the five interconnected storage pools as interim spent-fuel storage areas. After a Final Environmental Statement and a Safety Evaluation Report were issued in 1976, the licensing procedure was suspended. All auxiliary equipment (e.g., circulating water, unloading cranes) exists; and this facility could be ready to receive spent fuel once licensed. The current capacity is 400 metric tons of uranium (mtu). This capacity is limited only by the types of existing storage racks; however, employing the aforementioned racking techniques, a capacity of 1750 mtu can be achieved. A yearly handling rate (HR) indicates how much fuel can be unloaded from shipping casks and racked into the storage pools. In general, the HR is the limiting factor in the amount of fuel which can be stored in any given facility. For the BNFP, the handling rate at 1750 mtu capacity is 750 mtu/year.[8]

The GE-Morris facility is located in Goose Lake Township, Illinois as part of GE's Midwest Fuel Recovery Plant at Morris, Illinois. In 1973 the construction and testing of this facility as a reprocessing plant were completed. In addition, the appropriate licenses for storage of nuclear fuel were received. The facility, however, never operated as a reprocessing plant and now is licensed only for spent fuel storage. This license applies only for the GE owned facility. When and if the DOE purchases this facility for use as an AFR site, the license will have to be renegotiated. Initially the license permitted the storage of 100 mtu, but presently permits the storage of 750 mtu (based on Boiling Water Reactor, BWR, fuel only). If a mixture of forty percent BWR and sixty percent Presurized Water Reactor (PWR) fuel is stored, the capacity is approximately 700 mtu. Reracking existing pools would increase storage capacity to 1100 mtu. The DOE has proposed the construction of a totally new pool which would have a capacity of 1700 mtu, yielding a total capacity for Morris of 2800 mtu. Currently, approximately 350 mtu of spent fuel is stored at this location. This fuel has been received from the following reactor complexes: 1) Connecticut Yankee, 2) San Onofre, 3) Point Beach, 4) Dresden, and 5) LaCrosse.

The contracts were originally written with the intent to reprocess this spent fuel. Since then, the contracts have been modified for only the storage of the fuel. As of mid-1979, GE has suspended any further storage operations.[9] The handling rate, which depends upon factors such as design and degree of contamination of shipment casks, is between 200 and 300 mtu/year.[8]

The West Valley facility (WNYNSC) is located about fifty kilometers south of Buffalo in the town of Ashford, New York. This facility has the distinction of being the only facility in the United States not only licensed to reprocess spent nuclear fuel, but actually operating as a reprocessing faciltiy for a number of years. In addition, burial grounds for solid nuclear wastes are located on-site. The storage capacity can be increased via several methods. The present racks permit a storage capacity of 270 mtu. If a modified canister design is implemented, the storage capacity increases to 775 mtu. An upgrade to 1240 mtu is possible if multi-assembly storage canisters are locked to a supporting floor grid. Finally, a maximum capacity of 1500 mtu is achieved by the use of free-standing storage modules. In each scenario, the handling rate is 750 mtu/year. Presently 163.5 mtu of spent fuel is being stored at West Valley.[8] In 1978, NFS announced that the facility would no longer remain in the reprocessing business. In addition, the plant would no longer receive any spent fuel.[10] Presently, West Valley is in a shutdown condition.

New AFR facility designs are currently being studied. Present projections indicate that construction could start on the first new AFR in fiscal year 1985. Operation is expected in fiscal year 1990. The capacity of each new AFR is designed at 5000 mtu with a handling rate of 1000 mtu/year.[7] The AFR Program Plan calls for the construction of additional facilities on an as-needed basis. Since a licensing and construction period of five years is required, accurate projections of spent fuel storage requirements are mandatory.

THE APPROACH

Initial cost studies confirm the suspicion that interim storage facilities are desperately needed. An at-reactor storage pool can be reracked for $16,000 per mtu. Additional at-reactor pools cost on the order of $320,000 per mtu, whereas Federal AFR storage is estimated between $100,000 per mtu and $150,000 per mtu.[8] There exists a large economic incentive for the utilities to first rerack their existing on-site pools and then ship to an AFR facility. Thereupon, the demand for AFR storage space lies.

With time a formidable enemy in the battle to provide sufficient spent-fuel storage and a realization that incorrect decisions can cost into the billions of dollars, the director of the AFR Program requested some form of computer based tool with which to assist in the managerial processes.[11] This aid is a simulation technique applied to the back-end of the nuclear fuel cycle. In this section, an explanation of simulation and why simulation is chosen is presented. Next, the selection process involved in picking a particular simulation language is reviewed. Lastly, the selected language is explained.

Simulation

In a management process, four methods of decision aids are:

1) mathematical analysis,
2) experimentation,

3) experience and intuition, and

4) simulation.[12]

Mathematical analysis for a large, complex system is often intractable, and if not intractable, prohibitively expensive. Without simplifying assumptions, which can devalue a model, a mathematical analysis of the back-end of the fuel cycle is unrealistic due to the large number (approximately twenty) of interrelated variables (e.g., AFR facility on-line time, capacity, handling rate, individual reactor location, full core reserve). Similarly, experimentation on a prototype or model is not a viable option because of the expense and time consumption. A feasible management technique centers on experience and intuition. However, experience in storage techniques as applied to the present day fuel cycle is lacking. Intuition is closely interrelated with experience; generally, an intuitive decision results from some previous experience on a similar situation. The effects of intuitive decisions without a solid foundation of understanding or experience can prove hazardous to this country's health.

What is required is a tool, a technique which facilitates the actual decision-making process. Until recently, only in isolated applications has a computer been employed in conjunction with managerial expertise to improve the actual decision-making process.[13] There are many applications of computerized management information systems (MIS) in management. A MIS is a datafile, providing rapid data retrieval. However, a MIS only makes existing managerial processes quicker. A MIS does not improve upon the actual decision-making process itself.[13]

During the last decade, in conjunction with the improvement in computer technology, simulation methods have been developed which actually facilitate the decision-making process.[13] In the general sense, simulation means to model, or specifically, to study the behavior of a model.[14] A child playing with a doll-house is simulating the real world. Simulation provides the tool for predicting the dynamic characteristics of a system under observation, with the objective of gaining an understanding of the relationships among components of the system.[13] Here, the term system is used to represent an assemblage of interdependent physical (rather than abstract) objects that form a unified whole.[15] This technique actually improves the basis of the entire decision process.

The policies of the nuclear fuel cycle are complex, ever-changing, politicaly motivated, and technological in nature. Elections can drastically alter the course of investigations. Hence, the managers require a simulation technique which improves the decision-making methodology. If one relys on experience and intuition, the rapid changing environment can force the manager into a position of reacting to crises, rather than controlling activity.[13] Simulation provides the basis for a formal and efficient model which the DOE managers can apply to their decision-making policies: formal in the sense that the model can be precisely documented yielding an aid which can be quickly learned and applied to new situations, effective in the sense that costs do not rise in a linear proportion to the complexities of the system being modelled.[13]

Hence, simulation is deemed the best modelling technique to use on the back-end of the nuclear fuel cycle. This model provides a symbolic representation of the fuel cycle, allowing managers a mechanism to predict effects of alternate scenarios. A particular scenario can be examined in a matter of minutes, vastly superior to experimentation and/or mathematical techniques.

An exhaustive literature search is undertaken by two means. The first is a computerized examination of the energy information databanks at the Oak Ridge National

Laboratory. The second search method involves a hand review of various abstracts; not only the nuclear abstracts, but industrial engineering abstracts as well. Neither search produces any information directly related to the application of a simulation technique to the post-1977 philosophy of spent fuel disposition. Several endeavors, however, apply simulation to a fuel cycle which includes reprocessing. One of these works describes a simulation model of the production and inventories of the sequentially produced nuclear fuel, including a reprocessing feedback mechanism.[16] Another undertaking investigates the optimal amounts of stockpiled fuel materials via a mathematical model. The mathematical model is verified by a simulation technique.[17] Both of these efforts model the pre-1977 fuel cycle and neither considers any form of interim storage methodology. The articles do provide, however, general information on simulation techniques as applied to a fuel cycle. An international effort analyzes the use of Regional Nuclear Fuel Cycle Centres (RNFCC).[18] The advantages and disadvantages between a multination fuel center and individual national facilities are studied through various simulation models and submodels of the nuclear fuel cycle. Reprocessing plays a vital role in the RNFCC study.

There exists a definite void in material related to the present day fuel cycle, largely due to the newness of the task. The AFR Program was establish approximately two years ago. Only recently has sufficient material been accumulated with which to begin any work upon a simulation model.

Once simulation was chosen, a computer-based language had to be selected, the topic of the next section.

Language

In broad terms, a simulation language can be any computer language which can be employed in a simulation model. Examples of such languages include FORTRAN, BASIC, PL/I, and COBOL. However, a narrowing definition of a simulation language, and the one employed in this work, is a group of pre-packaged subroutines which a user can modify and employ in a simulation model. These subroutines contain certain functions common to most simulations.

In the beginning days of simulation, a realization was developed that different simulation models had various steps in common. The following list includes the common functions which almost every simulation model utilizes to some degree.[13] These features are

1) random number generation,
2) advancement of time,
3) recording data for output,
4) performing statistical analysis,
5) arranging outputs into specific formats, and
6) detecting and reporting inconsistencies and errors.

Endeavors began to provide simulation languages which eliminated the need for a user to reprogram these common functions. Although a complete listing of all the simulation languages would be prohibitively large; those which showed possibilities and were investigated in detail include: 1) SIMSCRIPT, 2) GASP IV, 3) CSMP, 4) GPSS, and 5) DYNAMO. Some of the factors one must consider in the selection process for any language include:

1) support by the computer facility,
2) ease of learning,
3) cost,
4) compiling and running time,

5) types of output, and
6) capacity for inserting user-written subroutines.[13]

The first criteria (i.e., supported by the computer facility) is usually the overriding criteria. At Virginia Tech, the five languages previously mentioned are supported. SIMSCRIPT is a complete language oriented toward event-to-event simulation. This language is probably the most powerful; however, the language is also one of the most complex and difficult to learn. Expert consultation is usually required due to the limited self-diagnostics.[12] GASP IV consists of a set of subroutines, coded in FORTRAN, which perform those functions listed above. If a user is familiar with FORTRAN, GASP IV can be implemented quickly. However, GASP IV is more restrictive than SIMSCRIPT in the size of certain characteristics of the system being modelled.[19] CSMP is a complete language useful in the solution of non-linear, integral-differential equations with continuous variables. GPSS is also a complete language oriented toward problems in which items pass through a series of processing and/or storage facilities.[12] GPSS is the easiest to learn; but, is one of the least flexible and becomes very slow and inefficient as the complexity of the problem increases.[20] Finally, DYNAMO is employed toward expressing micro-economic models of various firms by means of difference equations.

An initial review of the various languages eliminated both CSMP and DYNAMO as not applicable to modelling the back-end of the nuclear fuel cycle. The difficulty of GPSS to handle effectively the complexities which are involved in the fuel cycle model eliminated its usage. The two remaining languages, SIMSCRIPT and GASP IV, were further analyzed during the days of the final decision review. Both languages are poor in memory utilization, since neither permits dynamic allocation of memory. SIMSCRIPT is more difficult to learn, but GASP IV possesses greater limitations on the size of the problem to be modelled and the different output options. The final choice was based largely on selection factors two, three, four, and six. Either the limitations of GASP IV previously mentioned would not handicap the model, or the user could easily modify the GASP IV subroutines to overcome any such limits. An example would be the number of output plots available. GASP IV limits this value to ten; however, since all GASP IV subroutines are coded in

FORTRAN, the task of modifying the language to produce more plots is feasible. The features of cost, ease of learning, running time, and built-in diagnostics all favored GASP IV over SIMSCRIPT for the intended application.

One final observation is worth mentioning. Any simulation model can be produced by most any simulation language or any generalized computer language. The back-end of the fuel cycle could have been modelled using the computer language BASIC. The final choice is often one of personal preference. Due to the tremendous capabilities of present-day computers, factors such as compiling and running time become less of a selection criteria. A certain language may make the job easier since the user may possess an existing familiarity with said language. The computer facilities at Virginia Tech made the selection somewhat more difficult since numerous languages are supported. Many facilities possess only one, maybe two, simulation languages--making a selection process an inutile exercise.

GASP IV

GASP IV is a combined continuous-discrete FORTRAN-based simulation language. Discrete simulation occurs whenever changes to the dependent variables of a model occur at specific points during the simulated time. Continuous simulation permits continuous changing of the dependent variables over simulated time. GASP IV can operate in a discrete, continuous, or combined mode. This section overviews the features of GASP IV employed in the fuel cycle simulation model. For a detailed description of this language, the reader should refer to the references.[21-22]

This model exclusively utilizes the discrete mode of operation. Also, the model is simulated via the time function. In other words, a starting time is given and then GASP IV searches for the first happening or event in chronological order. Once finding such an event, action is taken depending upon what type of event. For example, if the event is a new storage facility coming on-line, the action would be to increase the total available spent fuel storage capacity. If the event is a reactor discharge, the action is to locate a storage facility with sufficient available capacity and simulate the storage of the discharged fuel. After the action is completed, time marches on until the next event, at which time the logic is repeated. This incrementing process continues until a halt is provided by the user.

There are four general functions provided by GASP IV that are of particular usefulness. These functions are 1) a filing system to store input data, 2) a time-advancing function, 3) a statistical gathering function, and 4) an output data gathering and reporting function. From previous discussions, realize that these functions could conceivably be user written, eliminating the need for any simulation language. However, this time consuming process adds little, if anything, to the value of the model.

The filing system utilizes a one-dimensional array. The size of this array is limited only by the computer facility. In this array, all input data are filed and tracked. A particular datum and its associated characteristics are referred to as an entry with attributes. An example of an entry is the add-on storage pool at the GE-Morris facility. Attributes include on-line time, capacity, and location. Another entry is a spent fuel discharge from a reactor. Its attributes are which reactor, which year, and amount of discharge. All such entries and associated attributes are filed into this single array with an extensive pointer system. This pointer system keeps track of each entry in chronological order.

The time-advancing function searches the event file for the next time event, removes the event (the entry and its associated attributes), and determines the action required. The particular action is coded by the user for each entry. This code is represented by the second attribute of an entry. In this model, a code of two implies that a storage facility is coming on-line and a code of one means that a reactor is discharging fuel. The time-advancing function interprets the code and routes the program to the appropriate subroutines.

The main statistic of interest is the percent utilization of each storage facility and repository. This value is the amount of spent fuel being stored at a particular facility divided by the total capacity of the facility. Each time spent fuel is stored or a change in capacity occurs, the utilization is calculated and the value stored in a GASP IV supplied subroutine. This subroutine keeps the data and compiles the mean, standard deviation, and maximum and minimum of the different values.

The simulation model provides two sources of output.

The first source is optional. If desired, the user can code a separate output subroutine which is called at the completion of the simulation. This routine can contain any additional user required output. Chapter Three describes the subroutine written specifically for this model. The second output is a GASP IV provided subroutine. This output consists of tabular and/or point plotable listings of the supply and demand for each storage facility and repository. This output is employed in the graphs depicting the spent fuel storage scenarios under observation. Examples are given in the following chapter.

In the form of a brief description, the functions provided by GASP IV, which are implemented into this model, have been reviewed. A more detailed examination of the role each function plays within the total model is given in Chapter Three.

Now that the need for a technique to aid DOE personnel is established and the specific methodology chosen to provide such a technique, the next phase involves a selection of sources of input data. The best model will yield erroneous results and prove worthless if the sources of input are outdated and unreliable. The final section in this chapter addresses the input data acquisition process.

THE INPUT

A realization that data covering all phases of the back-end of the nuclear fuel cycle are not centrally located was obtained in the initial phases of model development. Some data are on-hand in the Management Information System (MIS) developed by Virginia Tech for the Department of Energy's Savannah River Operations Office (SR). Other data have to be located elsewhere. This section explains what data are required and from what sources the data are obtained. The verification and updating of all the input is a complex undertaking. Measures to accomplish this are developed at Virginia Tech and elsewhere. Due to the importance of verification, a separate chapter examines the techniques used to perform such functions.

There exist three general categories of input data. The first is away-from-reactor (AFR) and at-reactor (AR) storage facility data; the second, storage repository data; and the third, spent fuel discharge data. An important definition is required: in this paper, discharge data refers to the amount of spent fuel that cannot be stored at the reactor (on-site) and not the amount of fuel out of the reactor core. The third category is further divided into three subcategories: 1) individual reactor discharge data by year from the present until 1999, 2) national discharge data by year from the year 2000 until 2020, and 3) foreign discharge data by year from the present until 2020.

AR and AFR Facility Data

The research and developmental studies to increase storage space for both AR and AFR storage pools and the studies related to the operation of AFR facilities are all under the cognizance of the AFR Program Office. A few examples of these studies include new methods of racking fuel assemblies, disassembly of fuel assemblies, dry storage concepts, prelicensing activities, and environmental impact statements. The results of these individual studies have both indirect and direct effects upon the fuel cycle model. A direct effect is an activity which, if delayed, can affect the on-line time or capacity of an AFR facility. An indirect effect is an activity which can alter the discharge data of a reactor (i.e., improved racking techniques would increase AR storage capacity and therefore lower discharge amounts). All such activities are maintained and tracked within the MIS designed and operated by Virginia Tech for the AFR Program Office.

The AFR MIS maintains current and complete information on all work packages authorized by the AFR Program Office. Each work package has a unique work breakdown structure number for reference and internal tracking. The data are divided into three sections: 1) general contract information, 2) financial data, and 3) activity data. The activity data are of prime importance to the operation of the simulation model. Figure 2 presents a simplified activity listing for the GE-Morris storage facility. Appendix A contains the complete listing for this work package. As is evident, the activities are characterized by a title, start date, end date, and predecessor and/or successor relations. This information is tracked and analyzed within the simulation model to determine the ultimate on-line time of that AFR facility. In addition, other data files contained within the MIS provide the capacities and handling rates of the various storage facilities. By including the AFR MIS

STATUS REPORT SPENT FUEL STORAGE 06/28/80 3- 2
8

ACTIVITY LIST:

WEN NO: 1213.
START: 01 OCT 80 END: INDEFINITE
TITLE: FACILITY IMPLEMENTATION-MORRIS

ACTIVITY: 1213.1
START: 01 OCT 81 END: 31 JAN 87 % COMPLETE 0
NAME: LICENSING

SUB-ACTY: 1213.1.1
START: 01 OCT 81 END: 02 OCT 81 % COMPLETE 0
NAME: CONSIDER LICENSE APPLICATION ACTIVITIES
AND MAKE APPLICATION FOR TRANSFER AND
RERACK
** PROGRAM CONTROLLED **

PREDECESSORS	SUCCESSORS
1212.1.1	1213.1.2
1211.3	
1121.4	

SUB-ACTY: 1213.1.2
START: 02 OCT 81 END: 30 SEPT 82 % COMPLETE 0
NAME: PREPARE TO RECEIVE LICENSE FOR TRANSFER
AND RERACK
** DELIVERABLE **

PREDECESSORS	SUCCESSORS
1213.1.1	1213.3.1

SUB-ACTY: 1213.1.3
START: 01 JULY 84 END: 01 JULY 84 % COMPLETE 0
NAME: SUBMIT APPLICATION FOR ADD-ON POOL
LICENSE
** PROGRAM CONTROLLED **

PREDECESSORS	SUCCESSORS
1213.1.2	1213.1.4
1212.1.2	
1142.3.6	

SUB-ACTY: 1213.1.4
START: 02 JULY 84 END: 31 JAN 87 % COMPLETE: 0
NAME: PREPARE TO RECEIVE LICENSE FOR ADD-ON
POOL
** CONTRACTOR MILESTONE **

PREDECESSORS	SUCCESSORS
1213.1.3	1213.3.2

ACTIVITY: 1213.2
START: 01 OCT 81 END: 31 JAN 87 % COMPLETE: 0
NAME: CONSTRUCTION DESIGN

SUB-ACTY: 1213.2.1
START: 01 OCT 81 END: 30 JUNE 82 % COMPLETE: 0
NAME: CONSIDER DESIGN PACKAGE AND COMPLETE
RERACK DESIGN
** CONTRACTOR MILESTONE **

PREDECESSORS	SUCCESSORS
1212.2.9	1213.3.1
1414.3	1213.2.2
1423.2.1	

SUB-ACTY: 1213.2.2
START: 01 ACT 84 END: 31 JAN 87 % COMPLETE: 0
NAME: CONDUCT ADD-ON POOL CONSTRUCTION DESIGN

PREDECESSORS	SUCCESSORS
1213.2.1	1213.3.2
1211.3	
1413.2	
1415.3.2	

ACTIVITY: 1213.3
START: 01 OCT 82 END: 26 FEB 89 % COMPLETE: 0
NAME: CONSTRUCTION

Figure 2. Activity Section of Work Package for GE-Morris

SUB-ACTY: 1213.3.1
START: 01 OCT 82 END: 30 JUNE 84 % COMPLETE: 0
NAME: PERFORM RERACK CONSTRUCTION
** CONTRACTOR MILESTONE **

PREDECESSORS	SUCCESSORS
1213.2.1	1213.4.3
1213.1.2	

SUB-ACTY: 1213.3.2
START: 01 FEB 87 END: 26 FEB 89 % COMPLETE: 0
NAME: PERFORM ADD-ON POOL CONSTRUCTION
** CONTRACTOR MILESTONE **

PREDECESSORS	SUCCESSORS
1213.1.4	1213.4.5
1213.2.2	

ACTIVITY: 1213.4
START: 01 JUL 84 END: INDEFINITE % COMPLETE: 0
NAME: OPERATION

SUB-ACTY: 1213.4.1
START: 01 JULY 84 END: 02 JULY 84 % COMPLETE: 0
NAME: START OPERATION WITH RERACK

PREDECESSORS	SUCCESSORS
1213.3.1	1213.4.4

SUB-ACTY: 1213.4.2
START: 03 JULY 84 END: INDEFINITE % COMPLETE: 0
NAME: OPERATE USING RERACK

PREDECESSORS
1213.4.3

COORDINATED
1213.4.2
1213.4.6

SUB-ACTY: 1213.4.3
START: 27 FEB 89 END: 28 FEB 89 % COMPLETE: 0
NAME: RECEIVE FUEL IN ADD-ON POOL

PREDECESSORS	SUCCESSORS
1213.3.2	1213.4.6

SUB-ACTY: 1213.4.4
START: 01 MAR 89 END: INDEFINITE % COMPLETE: 0
NAME: OPERATE FACILITY WITH ADD-ON POOL

PREDECESSORS
1213.4.5

Figure 2. (continued)

within the model, extreme flexibility and reliability are achieved in two ways. The first is that there exists only one place where such information resides. Therefore, when AFR managers update the MIS for purposes of reporting or planning, the input into the simulation model is also updated and no conflicts exist. Secondly, all activities within the MIS connected by predecessor/successor relations are trackable throughout the model. For example, if a manager desires to evaluate the effect of a one-year slippage of the site-specific environmental impact statement (EIS) for GE-Morris, the user simply changes the completion date for the EIS activity. A program tracks, through the predecessor/successor relationships, the effects of such a change. Slack times are accounted for and the on-line time for GE-Morris is changed.

An important digression needs to be presented at this point. Notice that the simulation model (i.e., computer) is not actually managing. The model merely provides a picture of the selected storage scenario. In the above example, if the manager is not satisfied with the outcome, his managerial responsibility would entail a shifting in resources (dollars and/or manpower) to avoid the effects of the one-year delay in the EIS. The manager is employing the model as an aid. He still retains the actual managerial duties.

Hence, a source for all the current data related to AFR storage facilities and AFR Program studies is obtained through the AFR Program MIS.

Repository Data

The responsibility for the design, construction, and operation of nuclear waste repositories falls under the Office of Nuclear Waste Isolation (ONWI). Currently, ONWI is initiating requests for proposals for the design and maintenance of a MIS similar to the AFR MIS. Therefore, the ability to track predecessor/successor related activities does not exist at this time. The data for on-line times, capacities, and handling rates employed in the simulation model come from a DOE document.[7] This source provides three scenarios for repository implementation. Each scenario is differentiated by the on-line time of the repositories. Table 3 gives the repository storage data for each of the three scenarios.

Table 3. Annual Repository Receipt Rate, MTU

Year Ending	Repository Startup Date 1997	2002	2006
1997	900		
1998	1800		
1999	1800		
2000	[a]2700		
2001	3600		
2002	5700	900	
2003	[b]8700	1800	
2004	9600	1800	
2005	11700	[a]2700	
2006	[c]14700	3600	900
2007	14600	5700	1800
2008	11700	[b]8700	1800
2009	13800	9600	[a]2700
2010	13800	11700	3600
2011	15900	[c]14700	5700
2012	18000	14600	[b]8700
2013	18000	11700	9600
2014	18000	13800	11700
2015	15000	13800	[c]14700
2016	12000	15900	14600
2017	12000	18000	11700
2018	9000	18000	13800
2019	6000	18000	13800
2020	6000	15000	15900

a. Second Repository Startup
b. Third Repository Startup
c. Fourth Repository Startup

Note: Handling rate for the first five years will be 1800 MTU/year; afterwards, 6000 MTU/year.

Discharge Data

Restating to avoid confusion, this is the definition of discharge data as used in this report: discharge data refer to the amount of spent nuclear fuel from a given reactor that is sent to an AFR facility and not the amount of spent fuel removed from the reactor core. This implies that improved AR storage techniques enhance AR storage and reduce the magnitude of the spent fuel discharges. With this understanding, the three subcategories of discharge data are reviewed.

Two sources of discharge data by individual reactor are examined. The first is the data produced by the Nuclear Assurance Corporation using their FUEL-TRAC futura system.[23] The second is the discharge data produced by the computer program DISFUL developed for the DOE by the S. M. Stoller Corporation.[24] A detailed discussion of each program and all the outputs is not intended in this paper; rather, a discussion of those features of each program which influence the final decision is presented. For a complete in-depth review of the programs, the reader is referred to the aforementioned references.

The initial investigation into these two programs shows remarkable similarities. Figures 3 and 4 are samples of output data from FUEL-TRAC and Figures 5 and 6 are outputs from DISFUL. The DISFUL data are current as of 1979; whereas, due to the accessibility of FUEL-TRAC data, the FUEL-TRAC figures are current as of 1977. This partially accounts for any discrepancies in the specific number of discharged assemblies between the outputs of the two programs. For example, Figure 4 shows that in 1993, fifty-two assemblies are to be shipped from Farley-1 to a repository. However, since the time that this FUEL-TRAC data was generated, the repository schedule has been delayed. The fifty-two assemblies would actually be sent to an AFR facility. This number compares favorably with the forty-six assemblies projected by the DISFUL program, as given in Figure 6. Additional factors, including at-reactor pool expansion estimates and power factors, are also partially responsible for the discrepancies between the outputs. The purpose of these figures is not in showing accurate discharge data; rather, to illustrate the type and format of the different outputs. Upon examination one can ascertain that either system can provide the required input data for the simulation model (i.e., reactor identification and amount of discharge fuel per year from present to 1999). Further investigation reveals differences in the methodologies used to arrive at

Assemblies Shipped

REACTOR	TYPE	TRUCK/RAIL	1977	1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000
FARLEY 1	P	T	-	-	-	-	-	-	-	-	-	-	2	-	-	-	-	-	-	-	-	-	-	-	-	-
FARLEY 2	P	T	-	-	-	-	-	-	-	-	-	-	-	-	2	-	-	-	-	-	-	-	-	-	-	-
PALO V. 1	P	R	-	-	-	-	-	-	-	-	-	-	1	81	80	80	81	80	-	-	-	1	80	80	81	
PALO V. 2	P	R	-	-	-	-	-	-	-	-	-	-	-	-	1	81	80	80	81	80	-	-	-	1	81	80
PALO V. 3	P	R	-	-	-	-	-	-	-	-	-	-	-	-	-	-	1	81	80	80	81	80	-	-	-	1
ARK. NUC. 1/1	P	R	-	-	-	-	-	-	-	35	56	56	-	-	-	-	-	-	35	56	56	-	-	-	-	-
ARK. NUC. 1/2	P	R	-	-	-	-	-	-	-	45	61	60	56	61	-	-	-	-	45	61	60	56	61	-	-	-
CALV. CLIFFS 1&2	P	T	-	-	-	-	-	12	145	145	144	145	73	-	-	-	12	145	145	144	145	-	-	-	-	12
PILGRIM 2	P	T	-	-	-	-	-	-	-	-	-	-	72	73	72	72	73	72	72	73	72	72	73	72	72	73
H. B. ROB. 2	P	R	153	-	-	-	53	52	53	52	53	-	38	52	53	52	53	52	53	52	53	-	38	52	53	52
BRUNSWICK 1	B	R	-	-	-	104	136	144	144	136	136	132	140	-	36	136	140	136	140	132	140	136	-	36	136	140
BRUNSWICK 2	B	R	-	-	-	124	144	144	140	140	136	140	-	-	116	140	140	140	140	140	140	-	-	116	140	140
S. HARRIS 1&4	P	R	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	38	53	52	53	52	-	-
S. HARRIS 2&3	P	R	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	27	53	52	53	52
ZIMMER 1	B	R	-	-	-	-	-	-	76	140	136	136	132	128	-	-	-	-	76	124	124	116	124	120	-	-
ZIMMER 2	B	R	-	-	-	-	-	-	-	-	-	-	-	-	-	-	196	196	196	196	196	-	-	-	-	196
PERRY 1&2	B	R	-	-	-	-	-	-	-	-	-	254	-	464	468	-	468	464	-	140	174	-	468	464	-	468
BRAIDWOOD 1&2	P	R	-	-	-	-	-	-	-	-	-	-	-	109	128	128	128	128	64	-	-	-	109	128	128	128
BYRON 1&2	P	R	-	-	-	-	-	-	-	-	-	-	110	128	128	128	128	64	-	-	-	110	128	128	128	128
LA SALLE 1&2 (D2)	B	R	-	-	-	-	-	-	191	232	192	-	-	-	-	-	88	215	216	192	-	-	-	-	-	176
LA SALLE 1&2 (D3)	B	R	-	-	-	-	11	272	129	212	212	-	-	-	-	60	212	-	212	212	-	-	-	-	124	212
LA SALLE 1&2 (QC)	B	R	-	-	-	-	-	-	84	388	200	-	-	-	-	-	-	-	320	12	-	-	-	-	-	-
LA SALLE 1&2 (LS)	B	R	-	-	-	-	296	496	376	392	392	392	-	335	504	396	400	392	392	372	396	286	263	300	392	400
INC. PT. 1	P	T	-	-	-	-	37	64	64	64	64	-	-	-	37	64	64	64	64	64	-	-	-	37	64	64
BIG ROCK PT.	B	R	26	22	-	18	22	22	22	22	22	8	22	1	18	22	22	22	22	22	8	22	1	18	22	22
MIDLAND 1&2	P	R	-	-	-	-	-	-	-	-	115	117	121	121	116	117	121	60	-	120	116	117	121	116	117	121
LA CROSSE	B	R	24	24	24	24	24	24	24	24	-	10	24	24	24	24	24	24	24	24	-	10	24	24	24	24
FERMI 2	B	R	-	-	-	-	-	-	270	200	192	216	200	192	200	200	196	200	200	192	200	200	196	200	200	192
GREENWOOD 2	P	R	-	-	-	-	-	-	-	-	-	63	68	69	68	69	68	69	68	69	62	69	68	69	68	69
GREENWOOD 3	P	R	-	-	-	-	-	-	-	-	-	-	-	63	68	69	68	69	68	69	68	69	62	69	68	69
OCONEE 1&2	P	T	-	-	-	-	-	-	-	-	-	-	-	108	112	112	112	112	112	112	105	101	108	112	112	112
OCONEE 3	P	T	-	-	-	-	-	-	-	-	-	-	-	56	56	56	56	56	56	56	-	50	56	56	56	56
McGUIRE 1	P	R	-	-	-	-	-	-	-	-	-	-	-	-	-	13	64	64	64	64	64	-	-	-	13	64
McGUIRE 2	P	R	-	-	-	-	-	-	-	-	-	-	-	-	-	13	64	64	64	64	64	64	-	-	13	64
CATAWBA 1	P	R	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	64	64	64	25	-	-	-	64	64
CATAWBA 2 (C2)	P	R	-	-	-	-	-	-	-	-	-	-	63	64	64	64	64	64	-	-	-	-	-	-	-	-
CATAWBA 2 (C1)	P	R	-	-	-	-	-	-	-	-	-	-	-	64	63	-	-	-	-	-	-	-	-	-	-	-

Figure 3. Sample Output from FUEL-TRAC Program

Reactor/Utility= Reactor # 1 (Farley-1)
MWe/Type/COD = 829/PWR
Transport Mode = Truck

Year	Pool Expansion Cost - $	Other Reactor	Assemblies Shipped To AFR	Repository
1977				
1978				
1979				
1980				
1981				
1982				
1983				
1984				
1985				
1986				
1987			2	
1988				52(1)
1989				52
1990				52
1991				52
1992				52
1993				52
1994				52
1995				52
1996				50
1997				52
1998				52
1999				52
2000				52

(1) Direct From Reactor

Figure 4. Sample Output from FUEL-TRAC Program

ANNUAL MTU SHIPPED

	REACTOR	UTILITY	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003
1	FARLEY-1	ALABAMA POWER CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	13	22
2	FARLEY-2	ALABAMA POWER CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5	PALO VERDE-1	ARIZONA PUB SERV CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	9	32	32	32	32	32
6	PALO VERDE-2	ARIZONA PUB SERV CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	9	32	32	32
7	PALO VERDE-3	ARIZONA PUB SERV CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	9	32	32
8	ARKANSAS NUCL ONE-1	ARKANSAS P AND L CO	0	0	0	0	0	0	18	21	21	21	21	21	21	21	21	21	21	21	21	21	21	21	21	21	21
9	ARKANSAS NUCL ONE-2	ARKANSAS P AND L CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5	23	23	23	23	23	23	23	23	23	23
10	CALVERT CLIFFS-1	BALTIMORE G AND E CO	0	0	0	0	0	0	0	0	0	0	23	44	44	44	44	44	44	44	44	44	44	44	44	44	44
11	CALVERT CLIFFS-2	BALTIMORE G AND E CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
12	PILGRIM-1	BOSTON EDISON CO	0	0	0	0	0	0	0	0	0	0	0	18	22	22	22	22	22	22	22	22	22	22	22	22	22
13	PILGRIM-2	BOSTON EDISON CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	28	30	30
14	ROBINSON-2	CAROLINA P AND L CO	0	14	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20
15	BRUNSWICK-2	CAROLINA P AND L CO	0	0	0	9	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27
16	BRUNSWICK-1	CAROLINA P AND L CO	0	0	26	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27
17	HARRIS-1	CAROLINA P AND L CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	7	46	46	46	46
18	HARRIS-4	CAROLINA P AND L CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
19	HARRIS-2	CAROLINA P AND L CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	30	46	46
20	HARRIS-3	CAROLINA P AND L CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
22	ZIMMER-1	CINCINNATI G AND E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	17	27	27	27	27	27	27	27	27	27
23	PERRY-1	CLEVELAND ELECTRIC	0	0	0	0	0	0	0	0	0	0	0	0	0	0	76	79	79	79	79	79	79	79	79	79	79
24	PERRY-2	CLEVELAND ELECTRIC	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
25	DRESDEN-1	COMMONWEALTH EDISON	4	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
26	DRESDEN-2	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	27	28	28	28	28	28	28	28
27	DRESDEN-3	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	7	28	28	28	28	28	28	28
28	QUAD CITIES-1	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	56	56	56	56	56	56	56	56
29	QUAD CITIES-2	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
30	ZION-1	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	23	54	54	54	54	54	54	54	54	54	54	54
31	ZION-2	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
32	LA SALLE-1	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	21	76	76	76	76	76	76	76	76	76	76	76	76	76
33	LA SALLE-2	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
34	BYRON-1	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	36	57	57	57	57	57	57	57	57	57
35	BYRON-2	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
36	BRAIDWOOD-1	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	36	57	57	57	57	57	57	57	57	57
37	BRAIDWOOD-2	COMMONWEALTH EDISON	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
38	CONNECTICUT YANKEE	CONN. YANKEE ATOMIC	0	0	0	0	0	0	0	0	0	0	0	5	23	23	23	23	23	23	23	23	23	23	23	23	23
40	INDIAN POINT-2	CONSOLIDATED EDISON	0	0	0	0	0	0	0	0	0	0	1	23	23	23	23	23	23	23	23	23	23	23	23	23	23
41	BIG ROCK POINT	CONSUMERS POWER CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3	3	3	3	3
42	PALISADES	CONSUMERS POWER CO	0	0	0	0	4	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27
43	MIDLAND-2	CONSUMERS POWER CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	34	40	40	40	40	40	40	40	40	40
44	MIDLAND-1	CONSUMERS POWER CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
45	LA CROSSE	DAIRYLAND POWER CO	0	0	0	0	0	0	0	0	0	0	0	2	3	3	3	3	3	3	3	3	3	3	3	3	3
46	FERMI-2	DETROIT EDISON CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	26	37	37	37	37	37	37	37	37	37
47	GREENWOOD-2	DETROIT EDISON CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	10	31
48	GREENWOOD-3	DETROIT EDISON CO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
49	OCONEE-1	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	39	48	48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
50	OCONEE-2	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
51	OCONEE-3	DUKE POWER COMPANY	77	24	0	0	0	0	0	17	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24	24
52	MCGUIRE-1	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	14	28	28	28	28	28	28
53	MCGUIRE-2	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	14	28	28	28	28
54	CATAWBA-1	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
55	CATAWBA-2	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
56	PERKINS-1	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
57	PERKINS-2	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
58	PERKINS-3	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
59	CHEROKEE-1	DUKE POWER COMPANY	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	11	32	32	32	32

Figure 5. Sample Output from DISFUL Program

PROJECTED ANNUAL SHIPMENTS FROM FARLEY-1 (829,MWE,PWR)

STARTUP 1978
OPERATOR ALABAMA POWER CO
NERC REGTON 6 (SERC)

FULL CORE SIZE (ASS.) 157
NORMAL YRLY DISCHARGE (ASS.) 46
ASSEMBLY WEIGHT, MTU 0.451

CONDITIONS
1. STORAGE PERMITTED WITHIN REACTORS DWN POOL ONLY
2. EXPANSION OF POOL ACCORDING TO CURRENT UTILITY PLANS
3. MAINTENANCE OF NORMAL YEARLY DISCHARGE CAPABILITY
4. NORMAL YEARLY DISCHARGE SIZE CALCULATED AT 70, PCT. P, F.

	POOL	TOTAL IN STORAGE		TOTAL SHIPMENTS		SHIPMENTS BY AGE					
YEAR	CAP.	ASS.	MTU	ASS.	MTU	NO.	AGE	NO.	AGE	NO.	AGE
1978	675	0	0.0	0	0.0	0	0				
1979	675	46	20.7	0	0.0	0	0				
1980	675	92	41.5	0	0.0	0	0				
1981	675	138	62.2	0	0.0	0	0				
1982	675	184	83.0	0	0.0	0	0				
1983	675	230	103.7	0	0.0	0	0				
1984	675	276	124.5	0	0.0	0	0				
1985	675	322	145.2	0	0.0	0	0				
1986	675	368	166.00	0	0.0	0	0				
1987	675	414	186.7	0	0.0	0	0				
1988	675	460	207.5	0	0.0	0	0				
1989	675	506	228.2	0	0.0	0	0				
1990	675	552	249.0	0	0.0	0	0				
1991	675	598	269.7	0	0.0	0	0				
1992	675	629	283.7	15	6.8	15	13				
1993	675	629	283.7	46	20.7	31	14	15	13		
1994	675	629	283.7	46	20.7	31	14	15	13		
1995	675	629	283.7	46	20.7	31	14	15	13		
1996	675	629	283.7	46	20.7	31	14	15	13		
1997	675	629	283.7	46	20.7	31	14	15	13		
1998	675	629	283.7	46	20.7	31	14	15	13		
1999	675	629	283.7	46	20.7	31	14	15	13		
2000	675	629	283.7	46	20.7	31	14	15	13		
2001	675	629	283.7	46	20.7	31	14	15	13		

Figure 6. Sample Output from DISFUL Program

the discharge data. Figure 7 depicts the information for both PWR and BWR fuel cycles used in the FUEL-TRAC futura system. These data are the basis for the calculations of discharge fuel for three United States energy forecasts. In turn, each forecast is evaluated under two cases, depending on AR storage time before shipment (ten years or 180 days). Figure 8 shows the basic options for the DISFUL program. Although specific options are different than those for FUEL-TRAC, a similarity exists. To this point of investigation, either data are acceptable.

The final selection is based on factors which are not actually related to the programs themselves. The DISFUL database is located on both the Babcock and Wilcox (B and W) computer and the Virginia Tech computer. S. M. Stoller employs the B and W computer facility for their use. The database at Virginia Tech is modified to operate under a generalized report writer system called MARK IV.[25] This system permits DOE officials to extract specific data from the database in a host of formats without employing the DISFUL program. For the simulation model, certain information contained within this database is invaluable. For example, the location of each reactor is given in longitude and latitude. These values are employed within the DISFUL program; however, they are not produced in any

BWR

Average MWe: 1000

First Cycle Length: 13.6 full power months

KgHM/ASBL: 183

	Initial Core	1st Reload	2nd Reload	3rd Reload	4th Reload
Fuel Loading					
MTHM**	125.6	36.0	30.5	27.3	28.5
Uranium Fraction	1	1	1	1	1
U-235 Enrich	1.87	2.755	2.755	2.755	2.755
Kg Pu-Fissle	0	0	0	0	0
Discharge					
W/O Initial Fuel		98.7	97.4	96.6	96.2
Uranium W/O		~100	~100	~100	~100
U-235 Enrich		.67	.726	.586	.761
Fissile Pu W/O Init.		.346	.484	.531	.578

PWR

Average MWe: 1000

First Cycle Length: 12.4 full power months

KgHM/ASBL: 462

	Initial Core	1st Reload	2nd Reload	3rd Reload	4th Reload
Fuel Loading					
MTHM**	78.4	26.1	26.1	26.1	26.1
Uranium Fraction	1	1	1	1	1
U-235 Enrich	2.60	3.25	3.25	3.25	3.25
Kg Pu-Fissile	0	0	0	0	0
Discharge					
W/O Initial Fuel		97.8	96.4	95.4	95.6
Uranium W/O		100	100	100	100
U-235 Enrich		.942	.781	.743	.918
Fissile Pu W/O Init.		.474	.604	.667	.660

* same for subsequent reloads.

** based on a 70% capacity factor.

Figure 7. FUEL-TRAC BWR/PWR Fuel Cycle Data

TYPES OF TRANS-SHIPMENT PERMITTED

1) Storage within reactors own pool only

2) Shipment within same or affiliated utility, system considered one unit

3) Shipment within same or affiliated utility, one intermediate site before federal facility

4) shipment to another utility within same NERC region

LEVELS OF POOL EXPANSION CONSIDERED

1) Expand pool according to licensed plans

2) Expand pool according to current utility plans (unlicensed expansions delayed 0 years)

3) Expand pool to maximum utility estimate in 1983

4) Expand pool to maximum SMSC estimate in 1983

REQUIREMENTS FOR OFF-SITE SHIPMENT CONSIDERED

1) Maintain full core discharge capability

2) Maintain normal discharge size capability

3) Ship assemblies stored 5 years post-irradiation

DETERMINATION OF NORMAL DISCHARGE SIZE

1) As estimated by utility

2) As calculated at 70. pct. capacity factor

Figure 8. Storage Options of the DISFUL Program

of the output. Since the database is on the computer at Virginia Tech, there is no problem in extracting the location of the various reactors and inputting this information into the simulation model. In addition, the procedure to link to the B and W computer and operate DISFUL is provided to Virginia Tech. An authorized user can run DISFUL and transfer the output to the Virginia Tech computer. These features of the Stoller System are basically free of any costs except for computer time.

In contrast, the FUEL-TRAC futura system is the property of the Nuclear Assurance Corporation. Neither the programs nor the databases are readily available; especially at no cost. Since both systems yield comparable data, the selection of DISFUL and its associated database is based on the tremendous ease of access and the no cost features of the Stoller System. In addition, Virginia Tech assists both DOE and Stoller in the maintenance of their database. The ability to control the accuracy of the input into the simulation model is reassuring. A detailed description of the method of assurance is given in Chapter Four.

The DISFUL program does not provide projections past the year 2003. The data just do not exist for individual predictions. For discharge data out to the year 2020, the values are total United States discharges by mtu per year. This data is given in Table 4.[26] Ten pseudo-reactors, arbitrarily located throughout the United States, are chosen to represent discharge points for these yearly estimates. This enables the model to function in a manner analogous to pre-2000. This approach is approved by DOE officals.[27]

The foreign discharge data are handled in a similar fashion. The data are by total mtu per year; hence, three pseudo-reactors are selected and the spent fuel equally divided. Two reactors are chosen on the East coast and one on the West coast. Once the fuel reaches the United States, said fuel is treated in a manner exactly like domestically produced fuel. Table 5 gives the anticipated yearly foreign discharge data.[28] Note that the total amount of foreign spent fuel is limited to 1000 mtu.

Now that the foundation for the model is complete, the next chapter discusses the actual model.

Table 4. Projected Reactor Discharges, MTU

Year Ending	Annual BWR	Annual PWR	Annual Total
2000	1925	3090	5015
2001	1685	3375	5060
2002	1760	3520	5280
2003	1840	3680	5520
2004	1910	3820	5730
2005	1990	3980	5970
2006	2060	4130	6190
2007	2140	4280	6420
2008	2210	4420	6630
2009	2290	4590	6880
2010	2360	4720	7080
2011	2435	4875	7310
2012	2490	4980	7470
2013	2540	5080	7620
2014	2590	5180	7770
2015	2630	5260	7890
2016	2680	5360	8040
2017	2730	5470	8200
2018	2780	5570	8350
2019	2830	5670	8500
2020	2880	5770	8650

Table 5. AFR Program Foreign Storage Requirements

Year Ending	*Storage Required, (MTU) Cummulative
1979	0
1980	0
1981	50
1982	100
1983	220
1984	340
1985	435
1986	575
1987	690
1988	885
1989 and beyond	1000

*DOE is proposing to provide storage for a maximum of 1,000 MTU foreign fuel.

CHAPTER THREE - MODEL

This chapter describes the simulation model of the back-end of the nuclear fuel cycle. Through the use of flow charts and graphs a complete understanding, including all options, is presented. Due to the complexities of the interrelated subroutines, extensive effort is exerted to ensure that the subroutines function as desired. In this chapter, emphasis is placed on the techniques employed to ensure the accuracy of the various subroutines. The verification of the model as a unit is presented as a separate section at the end of this chapter. Appendix B details the steps in the model verification process and includes sample output.

Those functions common to most simulation models are handled by GASP IV subroutines as reviewed in Chapter Two. Except where required for clarification and enhancement a further description of these routines is not given in this work; the reader is referred to the references.[21-22] These references provide a much more complete and authoritative description than can be provided in this work. The GASP IV supplied subroutines provide supportive functions; whereas, the user supplied subroutines provide the modelling.

THE SIMULATION MODEL

Figure 9 presents the overall flow through the simulation model. The model is divided into two systems with each system composed of a series of modules. The delineation between the systems centers upon those computer programs independent of control by GASP IV subroutines and those programs orchestrated by GASP IV. This demarkation also separates those functions which are performed on one type of computer and those performed on another type of computer. System One runs under IBM's Conversational Monitoring System (CMS) language. This is an interactive language permitting active user participation. System Two runs on the computer which accepts batch jobs only, providing greater central processing unit (CPU) utilization. In addition, the GASP IV subroutines are available only on the batch CPU. However, to the user no distinction is evident since executive programs are developed which automatically submit System One's output, along with the modules of System Two, to the batch machine. These executive

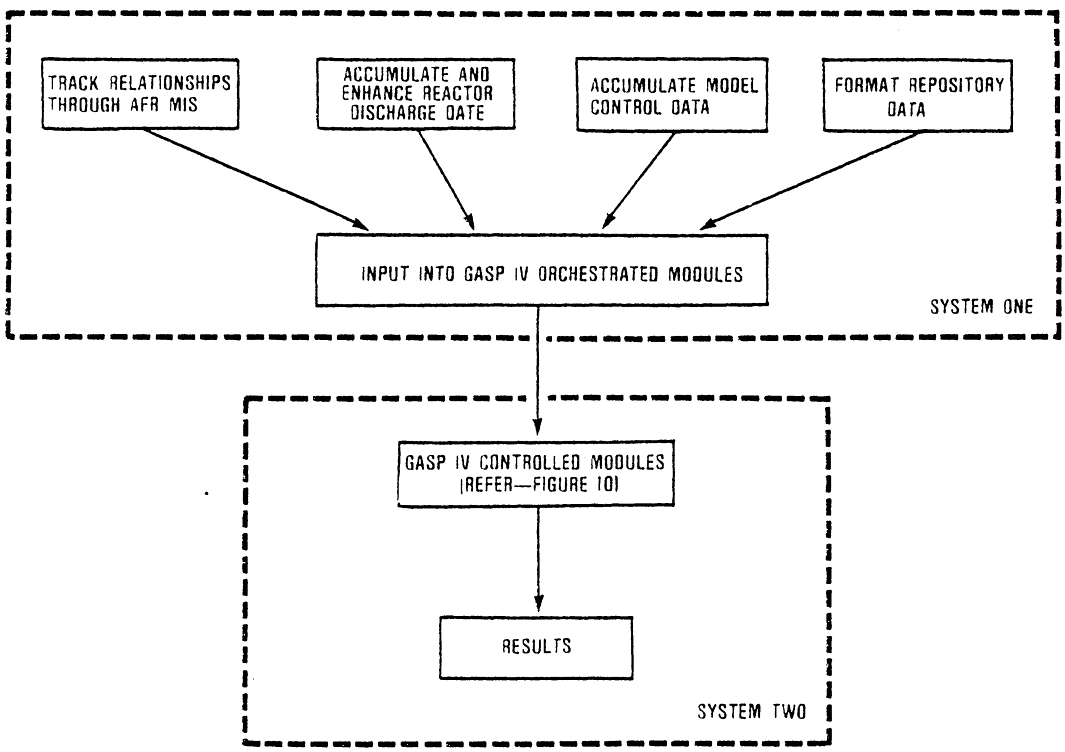


Figure 9. Flow Diagram of the Back-end Nuclear Fuel Cycle Simulation Model

programs are custom designed for this simulation model. The programs ensure the proper operation of the model by different managers possessing varying degrees of computer savvy. In turn, the output of the entire simulation model is returned to the CMS computer for user scrutiny. The employment of the two computers provides for the most flexible and efficient model possible. The interactive features enhance flexibility and the batch CPU enhances efficiency. Both systems are totally interrelated, unitizing the simulation model.

System One is described first, then the modules comprising System Two. Each module within the systems is examined, both as an individual entity and as the module relates to the entire model. Additional figures elaborate System Two. The reader is directed to refer to the appropriate flow diagram throughout the discussion of the simulation model.

System One

The modules making up System One, in many respects, are the most vital to the simulation model. These modules are responsible for the correctness of all input data.

The precision of this system lays a solid foundation for System Two.

To ensure such precision, features are designed into the model which minimize the occurrence of various errors. There are two classes of errors: 1) errors which cause an abnormal termination of the model and 2) errors which do not terminate the programs but alter the correctness of the output. Of these classes, the later is by far the most serious and hardest to detect. An abnormal termination is spotted immediately; all sorts of visual hints are present in the output. Although the cause of the termination may not be obvious, the user has no doubts that presented output is worthless. Since no error messages are printed, the other type of error can often go unnoticed for a long period of time. Due to the massive amount of required input and the interrelations amongst said input, an error of this nature may not be obvious to the user. For example, the output may report that reactor number twenty cannot ship spent fuel in 1982, when in fact the actual date is 1984. One cause for errors of this classification is the format of the input variables. Most input is formatted as integer and must be right justified. Whenever this right justification is lacking, the computer pads the number with zeros, changing its value. One example deals with the random number generator seed. The seed must be odd; however, if not correctly justified the seed is padded with zeros and becomes a even integer. The random number generator still spews out numbers, but the randomness is lost.

The first class of errors is remedied by dogged programming determination. The second class is corrected by extensive use of user query programs which extract the input data from the appropriate sources. These programs are designed by the author as an integral part of the simulation. The programs accept free-format input and then automatically format the data into the appropriate columns. In addition, echo statements provide the user with a look at the value just inputted. If an incorrect value is visualized, another chance is given. Internal range checks ensure ball-park accuracy. For example, if a repository on-line time is keyed as 1977, instead of 1997, the program politely informs the user that 1977 is past and no repository has yet to become operative.

Each module is now reviewed, commencing with the module which works upon the AFR Management Information System (MIS).

AFR MIS Module

Chapter Two examines the AFR MIS in detail and provides sample output which illustrates the type of data contained within. A brief refresher of the MIS is given here.

The AFR MIS contains every work package (e.g., contract) awarded by the AFR Program Office at Savannah River. Each work package is further divided into three sections: 1) general information and scope of work, 2) financial data, and 3) activity data. Figure 2 shows an activity breakdown for a work package centering on the GE-Morris facility implementation. Appendix A gives a complete listing of this work package. An activity is defined to be a specific unit of work within the work package characterized by a brief description of the work, the start date, the end date, and any predecessor and/or successor related activities. Each work package plays a critical role in the realization of the AFR Program objective (i.e., provide adequate storage space). The activities are the critical path elements forming a network which is necessary in the evaluation of the progress of the AFR Program. The AFR MIS module acts upon all the activities and their interrelationships contained within the AFR MIS.

The working of the AFR MIS module is as follows: A manager updates or changes a specific activity under his jurisdiction. This may involve the changing of the end date of an activity or changing the predecessor relationship of an activity. After this and any other changes are made to the entire MIS, the tracking of the activities commences. An interrelationship tracking program provides two options. The first option permits the program to automatically adjust all related predecessor/successor dates; the second, flags discrepancies. For example, if the change is to delay the site selection for the new AFR facility, then the user can specify that all successor activities be automatically delayed the required amount. Slack times act as buffers. In this example, successors would include site design, construction, and operation. The second option flags conflicts and/or logic failures, without affecting dates, permitting the user the choice of adjusting the activities in whatever manner desired. A one year delay in site design may be unacceptable; hence, the user can simulate increased resources acting on design by placing the delay at only six months. Already these preliminary results are providing managerial aids.

After the activities of the MIS are tracked and adjusted in the desired manner, the milestone extraction program is invoked. This program automatically retrieves the culminating activities of each AFR faciltiy. These activities represent the operation of the facilities. In addition, the required characteristics of the facilities are extracted. The characteristics include 1) the on-line time, 2) the handling rate, 3) the storage capacity, and 4) the amount of on-site spent fuel. All this information is correctly formatted and transferred into the input module.

In summary, the AFR MIS module initiates the simulation model by tracking all the events which ultimately affect the operation and capacities of the AFR facilities. Activities, such as site selection, affect the on-line times and activities, such as studies to improve storage fuel density, affect the capacities. Further, all data which are required to interface with the other modules of the simulation model are extracted, formatted, and forwarded to the appropriate module. Intermediate results can be selected which provide the user with a mechanism to check and/or correct a desired scenario.

Manual tracking provides the basis for verification of this module. An end date of an activity is changed and the effects manually tracked throughout the entire MIS. These results are compared with the intermediate output of the tracking program, confirming proper operation. In a comparable fashion, the milestone extraction program is guaranteed.

Reactor Discharge Data Module

This module accumulates the spent fuel discharge data originating from the DISFUL program and DOE documents. The DISFUL program provides spent fuel discharge data by individual domestic reactor for the period from 1979 through 2003; however, only the data through 1999 is used in this simulation model (this decision is determined by AFR Program officials). The DOE documents yield annual United States estimates of spent fuel discharges from the year 2000 to 2020 and foreign discharge estimates from 1979 until 2020. A detailed review of these documents is given in Chapter Two.

The foreign fuel data from the year 1979 to the year 2020 and the domestic discharge data for the year 2000 and beyond are handled in an analogous way. Three pseudo-reactors are allocated the foreign fuel allotments. Two of these reactors are located on the East coast; the remaining, the West coast. The data for the years 2000 and on are divided amoung ten pseudo-reactors arbitrarily located throughout the United States. This method of assigning national estimates to pseudo-reactors is required since the majority of discharge data, which originates from the DISFUL program, is listed by individual reactor. All discharge data can then be handled by the same programming techniques.

As mentioned, the output of the DISFUL program is broken down by individual reactor. Several desired simulation model features require additions to the DISFUL output. These features include

1) storing fuel against the full core reserve (FCR) capacity of an individual reactor,
2) determining the distance between a reactor and each AFR facility and repository in order to decide shipment priority, and
3) combining reactor discharges according to state.

An auxiliary program massages the output of DISFUL to include the latitude and longitude, state code, and FCR capacity of each reactor. Some data are extracted directly from the Stoller Database, while other data (e.g., state codes) are retrieved from auxiliary files created specifically for the model.

After the DISFUL output is formatted, these data are appended with the information on foreign discharges and on future discharges (i.e., beyond 1999). The reason for using pseudo-reactors now becomes apparent since at this point all discharge data are formatted identically for handling by the GASP IV routines.

Temporary write statements injected into the module provide output which is utilized in checking for proper operation. Once precise operation is confirmed by hand calculations these temporary statements are removed.

Model Control Data Module

The simulation model possesses options which permit a user to tailor a specific storage scenario to his wishes. Table 6 gives a functional description of these options. Each option is explained in detail during the discussion of its associated module. This module queries the user for the desired information, providing clarification when

Table 6. Simulation Model Control Options

Option	Function
Starting year	Sets the beginning time for the simulation
AFR shipment priority	Determines the shipment method between a reactor and an AFR facility (e.g. ship fuel to the nearest AFR site)
Repository shipment priority	As above, except for shipment from a reactor to a geologic repository
Full core reserve	If invoked; permits storage in the FCR section of the at-reactor storage pools
Temporary file storage	If invoked; tracks, by reactor, the amount of fuel which cannot be stored either at-reactor or away-from-reactor
Detailed output	If invoked; the output includes a listing of each discharge and the originating facility
Accident analysis	If invoked; simulates the possibility of an undesirable event causing a delay in fuel discharges

requested. As explained previously, by employing such a query program, format errors are avoided. The user inputs a specific value in free-format and then the program assumes the responsibility to format the variable in the needed manner. The logic of System Two is controlled by the variables chosen by the user through this module.

Repository Data Module

The specific data covering repository operation times and capacities are given in Chapter Two. Therefore, the module simply extracts the information from a disk file and transfers said data into the input module. However, functions have been programmed into the simulation model in anticipation of more detailed information concerning all aspects of repository design, construction, and operation.

The author has met and exchanged ideas with ONWI officials. As explained in Chapter Two, ONWI has the responsibility of overseeing the efforts involved in the development and operation of all federal nuclear waste storage repositories. In the fall of 1980, ONWI will commence the development of a MIS along similar lines as the AFR MIS. As this information system is nurtured into a complete MIS including activities and interrelationships, the author anticipates using the tracking and milestone programs developed for the AFR MIS against the ONWI MIS. Format and other minor modifications are to be expected; however, the heart of the programs will remain. Chapter Seven, covering recommendations, expounds upon this concept.

Input Module

The input module has the non-glorified but extremely important task of organizing the outputs from the various modules of System One. Once assembled, these data are combined with the modules of System Two and submitted to the batch computer at Virginia Tech. Once on the batch machine, the routines interface with the GASP IV subroutines and continue the simulation.

System One Reflections

This system probably has the greatest role to play within the simulation model. A series of modules track through all the data associated with the back-end of the nuclear fuel cycle. The modules are responsible for ensuring not only the mechanical but the logical correctness of said data. All interrelated activities within the AFR MIS are examined, slack times accounted for, and required knowledge extracted. Reactor discharge and repository facts from the most current sources are employed to guarantee the most reliable input into System Two. Finally, several auxiliary programs query the user and set the control mechanisms for the entire model. These programs pratically eliminate input mistakes common to other models.

Each module is separately tested. After all are individually certified, the modules are combined to form System One. Once again hand calculations invoking sample data prove the interrelated workings of this System.

System Two

System Two accepts the lore amassed by System One. Figure 10 provides a flow diagram of System Two. The first block, labelled INPUT, represents the input into

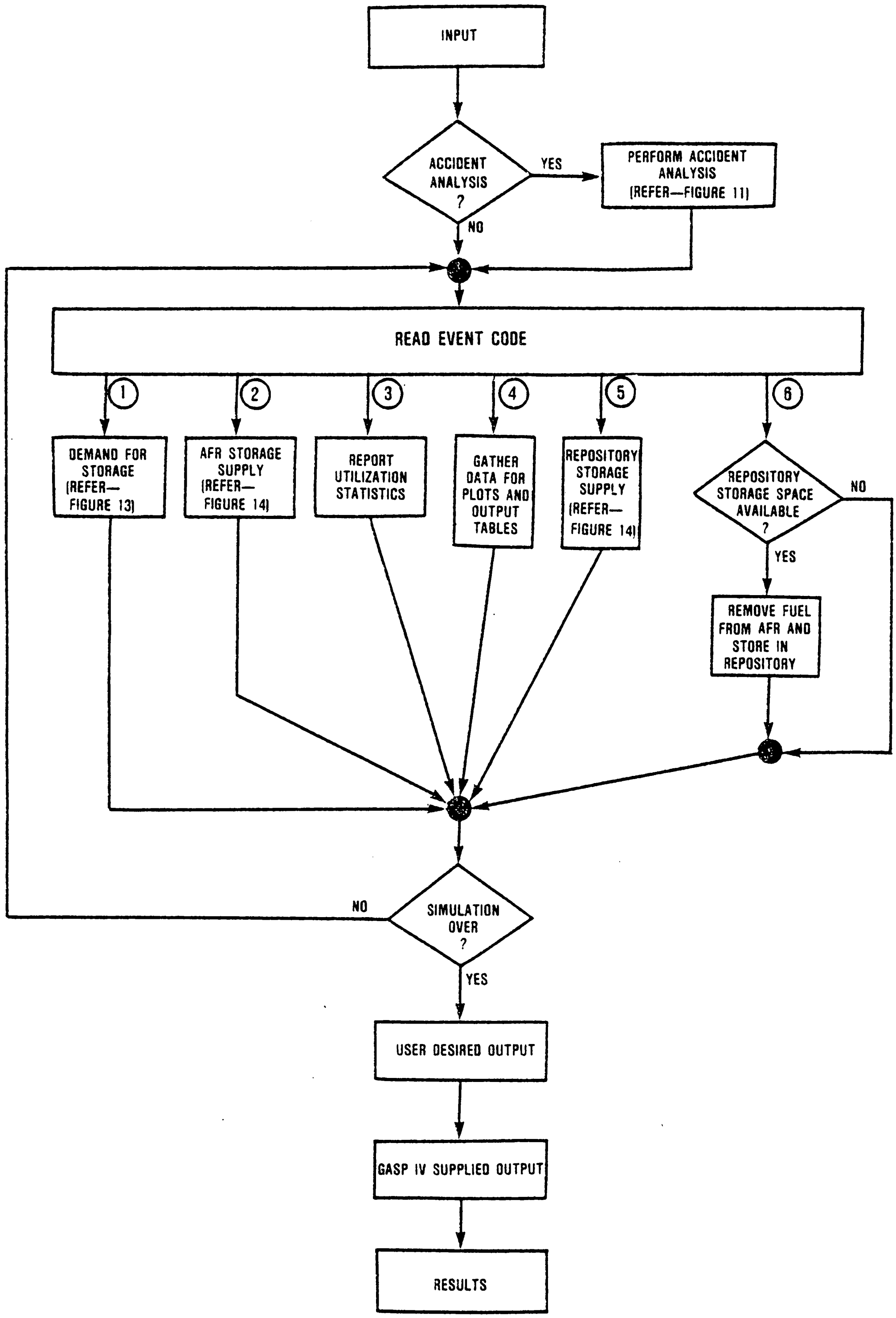


Figure 10. System Two Flow Diagram

this System accumulated by the modules of System One. Before GASP IV receives functional control, an optional accident analysis module massages the output from System One. The remaining modules of the model function under the guidance of the standardized GASP IV routines. Each module is discussed as sequenced in the flow diagrams.

Accident Analysis Module

The procedure for satisfying the spent fuel storage requirements depends not only on technological issues, but on governmental policies. However, these governmental policies are dictated by political elections, lobbying groups, law suits, nuclear and related accidents, and a multitude of other occurrences. Each potential effect has an amount of uncertainty associated with its probability of happening. Hence, the DOE managers are faced with making decisions under uncertainty.

Uncertainty is defined as a state of knowledge in which the probability that each event actually occurs is either not known or is not based on a meaningful statistical base.[29] A manager must make a decision in an environment of incomplete knowledge, being forced to use personal (subjective) judgments. There are three classes of uncertainty:

1) complete knowledge where each future event occurs with some degree of chance,
2) complete ignorance where neither subjective nor objective knowledge of the probabilities of future events exists, and
3) partial ignorance where several probabilities of future events are either missing or incomplete.[29]

Under complete ignorance the user may use whatever rational decision criteria he desires to arrive at a final choice.[29]

A mechanism to account for the uncertainties associated with the nuclear fuel cycle is programmed into the simulation model. This module is called the accident analysis routine, where the term accident is used in its broadest sense. That is, an accident refers not only to a nuclear accident, but any unintentional or undesirable happening. The user has the option whether or not to exercise this module. Although most uncertainty can be categorized by partial ignorance, the uncertainties associated with the back-end of the fuel cycle largely exist under the second class; namely, complete ignorance.

The presidential election of 1976 resulted in a redefinition of the nuclear fuel cycle. Certain candidates for the 1980 elections want to terminate nuclear power altogether; others, increase its use. There is just no subjective or intelligent method to predict probabilities associated with political directions--a vital influence on the nuclear industry.

The accident analysis module affects the discharge data only. The AFR facility and repository on-line times and capacities are not altered. The author reasons that an undesirable happening has a much greater probability of shutting down nuclear reactors than stopping construction of storage facilities. And shutting down reactors affects discharge data. In fact, an undesirable event might just move up the timetable for AFR facility operation in order to safely store spent fuel. The sensitivity toward an undesirable event (accident) and results of the simulation model with the accident analysis module exercised are given in Chaper Five. This current section explains how the accident parameters are calculated and then how the discharge data are adjusted. In this adjustment, the two required parameters are the number of years that the discharges are to be delayed and the percentage of reactors affected by the accident.

First, an overview on the workings of the accident module is given, followed by a detailed description. When employing the accident analysis module, a user selects a variety of parameters including the yearly accident rate, the maximum number of years to delay discharges in the worst case accident, and the coefficients of the various equations used in the module. These coefficients permit the user to tailor the analysis to his beliefs. The values are utilized to determine the number of years to delay the discharged spent fuel and the percentage of reactors affected by the simulated accident. Finally, the individual discharges are adjusted the appropriate amount.

Figure 11 is the detailed flow diagram for the accident analysis module and Figure 12 graphically depicts the series of calculations performed in determining the number of years of delay in the discharges of spent fuel and the percentage of the reactors affected for any given year. Noted on Figure 12 are all user inputted variables. The reader should refer to both figures during the following explanation.

The analysis is performed yearly, until the time exceeds the value inputted by the user. A random number between zero and one is obtained, multiplied by one hundred, and compared with the yearly accident rate. If

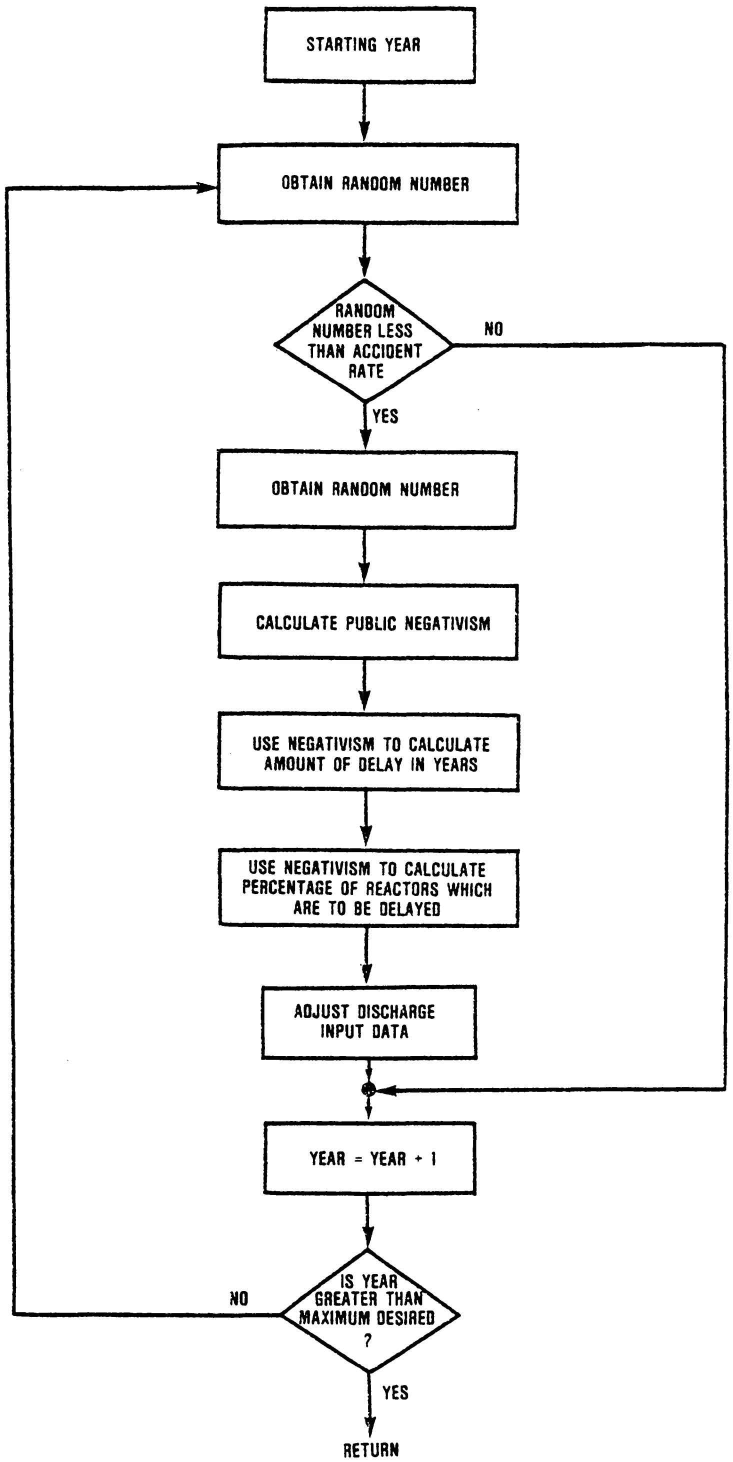
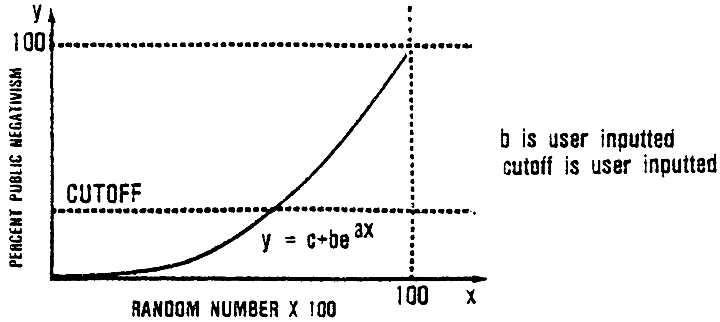
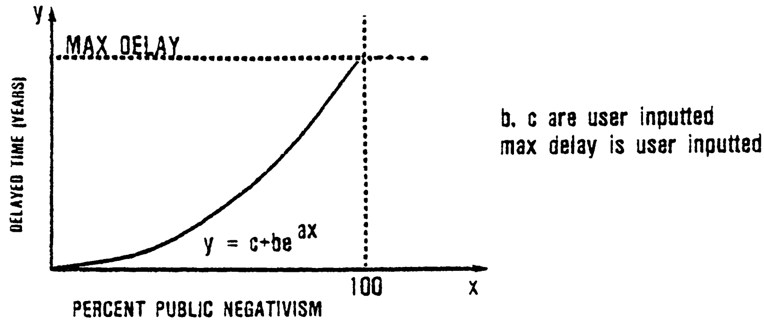


Figure 11. Accident Analysis Module of System Two

CALCULATION
ONE



CALCULATION
TWO



CALCULATION
THREE

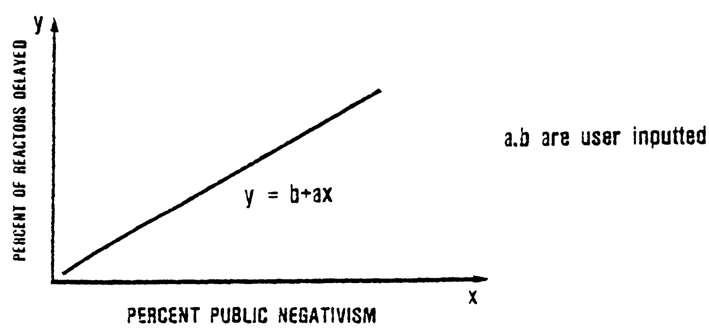


Figure 12. Relationships between Variables in Accident Analysis Calculations

the first random value is greater than the accident rate no analysis for that year is performed. The year is incremented by one and compared to the analysis termination time. If this time is exceeded, the entire accident analysis ends and control is transferred back to the GASP IV subroutines. However, if the first random variable is less than the accident rate, a series of calculations are performed, as illustrated in Figure 12.

The first calculation involves choosing a second random number. The first random number merely confirms or denies the existence of an accident, any accident of any severity. This second random number, in conjunction with the exponential equation, determines the severity of the accident. The severity is limned as a percentage increase in public negativism. The general form of the exponential equation is

$$y=c+b*\exp(a*x)$$

where the boundary conditions are

1) when x=100.0, y=100.0 and
2) when x=0.0, y=0.0.

The user influences the relationship between the second random number and the value of public negativism by the selection of the constant b. The values of a and c are calculated by the application of the boundary conditions. Once the percentage of negativism is calculated, this percentage is compared with the cutoff value (also user inputted). The cutoff furnishes the user with a mechanism to place a threshold value on negativism, below which no delays in spent fuel discharges can occur.

Whenever the negativism is above the cutoff value, the delay time and percentage of reactors affected are determined by two separate calculations. The second deciphering of the analysis uses the exponential equation

$$y=c+b*\exp(a*x)$$

and the boundary condition such that when x equals one hundred, y equals the maximum delay as selected by the user. In this equation, both constants b and c are user selected and a is determined by the boundary condition. The variable x is the value of negativism determined in the first calculation; the value of y, the number of years to delay reactor discharges.

The percent of reactors which are to be delayed is calculated thirdly by the equation

$$y=b+a*x$$

where both constants a and b are user selected. Here the value of x is equal to the negativism determined in the first calculation and the value of y is the percentage of reactors affected by the accident.

Given the two values, delay time and percentage of reactors affected, a subroutine adjusts the reactor discharges. One by one, each reactor discharge is examined over the period starting with the current year under analysis and continuing into the future for a number of years equal to the calculated delay. As each reactor discharge is examined, a random number is retrieved which is compared with the percentage of reactors affected (third calculation). If this random number is less than the percentage, then the reactor discharge is delayed the corresponding number of years. The amount of delay depends upon the year. For example, consider the year under analysis (base year) as 1981 and the delay time (second calculation) as three years. If reactor A has no discharge in 1981, but has a discharge in 1982, then the discharge in 1982 is moved back only two years. If reactor B has a discharge in 1981, that discharge is moved back a full three years. A temporary file accounts for the effects upon each reactor to ensure that redundancy does not occur. Once a reactor has its discharge adjusted, said reactor is exempt from further adjustments until the base year is incremented. In the previous example, reactor B is exempt after 1981. This guarantees that when discharges in 1982 are checked, reactor B does not have its discharge moved back an additional two years. Remember, throughout this example using reactors A and B the base year is 1981. After all adjustments are made for 1981, 1982, and 1983 (since the delay time is calculated at three years), the base year is incremented by one year and new values of public negativism, delay time, and percentage of reactors affected are calculated.

The preceding procedure is repeated once a year until the base year exceeds the limit as set by the user. At this point all reactor discharges of spent fuel have been adjusted according to the parameters of the accident analysis module.

Ensuring that this accident module performs as outlined involves a series of measures. The first is checking each subroutine comprising this module on an individual basis. Before each subroutine is combined into the simulation model, the subroutines are fitted into separate stand-alone programs. Thus a testing FORTRAN program is utilized to check the random number generator; a testing program is utilized to check the calculations for negativism, delay time, and percentage of reactors affected by the simulated accident; and so on until each component is certified. Next, the analysis module is compiled in its entirety and verified by using a sampling of discharge data. GASP IV provides an echo check on the main file containing all the events before the incrementation of time. By running the simulation model with and without accident analysis, the echo check can be employed, along with numerous programmer inserted write statements, to confirm the proper working of the analysis. The temporary write statements cause an outputting of the values of the random numbers, delay times, and other pertinent values. The author then performs hand calculations and compares the accident analysis adjusted discharge with these hand calculations and the non-adjusted data. All checks verify the accuracy of the accident analysis module.

The accident module does not have to be invoked--the decision rests with the user. After the analysis is performed, the input data are arranged into chronological order via a file pointer system. Now the actual simulation of the passage of time begins by extracting the first entry from the main GASP IV file (event file). Interpretation of this entry directs the action of the simulation. The specific action is a function of the user selected event code, as discussed in the following section.

Event Code Module

An event is an occurrence which affects the status of the system being modelled. The effect upon the system can be no effect. For example, the event could be a decision which has as one option the choice to leave the status of the system as is. In GASP IV, the possible events are categorized by the mechanism by which they are scheduled. Those events which occur at a specified projected point in time are time-events. Those that occur whenever the system reaches a particular state are referred to as state-events.[21] In the fuel cycle simulation model, all events are classified as time-events. This method of simulation is often called "next event" simulation.

An example of an event is a reactor discharge of spent fuel. This event is characterized by a series of attributes. In this example, attributes include reactor identification, year of discharge, and amount of discharge in metric tons of uranium (mtu). An entry is a specific happening and falls within an event classification. An example of an entry is reactor 25 discharging fifty mtu of spent fuel in 1984. The first two attributes of each entry (and hence each event) dictate the action to be considered. The first attribute is the ranking attribute. In this model the ranking attribute represents time. The second attribute is termed the event code and dictates the course of the simulation. Table 7 lists each event code employed in this simulation and associated function. Appendix C details the events and all the attributes.

GASP IV scans the event file, extracts the first entry, and examines the second attribute. This attribute pilots the simulation to the appropriate subroutines. The actions associated with each event code are perlustrated in numerical order, starting with event code one--storage demand.

Table 7. Simulation Model Events and Associated Functions

Event Code	Function
One	Discharge spent fuel from a specific reactor.
Two	Change the capacity and/or handling rate of an AFR facility.
Three	Report the utilization statistics (e.g. mean, standard deviation) for each AFR facility and repository.
Four	Accumulate storage supply and demand data for each AFR facility and repository (to be used in final output plots).
Five	Change the capacity and/or handling rate of a repository facility.
Six	Remove fuel from the AFR facilities and store in the repositories.

Storage Demand Module

An event code of one indicates that a certain amount of spent fuel is required to be stored in either an AFR facility or repository. Figure 13 represents the flow of actions required to simulate spent fuel storage. The first task is to ascertain which shipment priority between reactor and storage location is desired. There exists five different shipment options. The user selects the desired option for shipment from the reactor to the AFR facility and from the reactor to the repository. The options do not have to be the same for AFR facility and repository. The five shipment options are:

1) nearest location, and if unavailable then the next nearest location;
2) lowest utilized location, and if unavailable the next lowest utilized;
3) nearest location, and if unavailable the location with the lowest utilization;
4) lowest utilized location, and if unavailable the nearest location; and
5) user specified location (e.g., reactor number ten ships to GE-Morris).

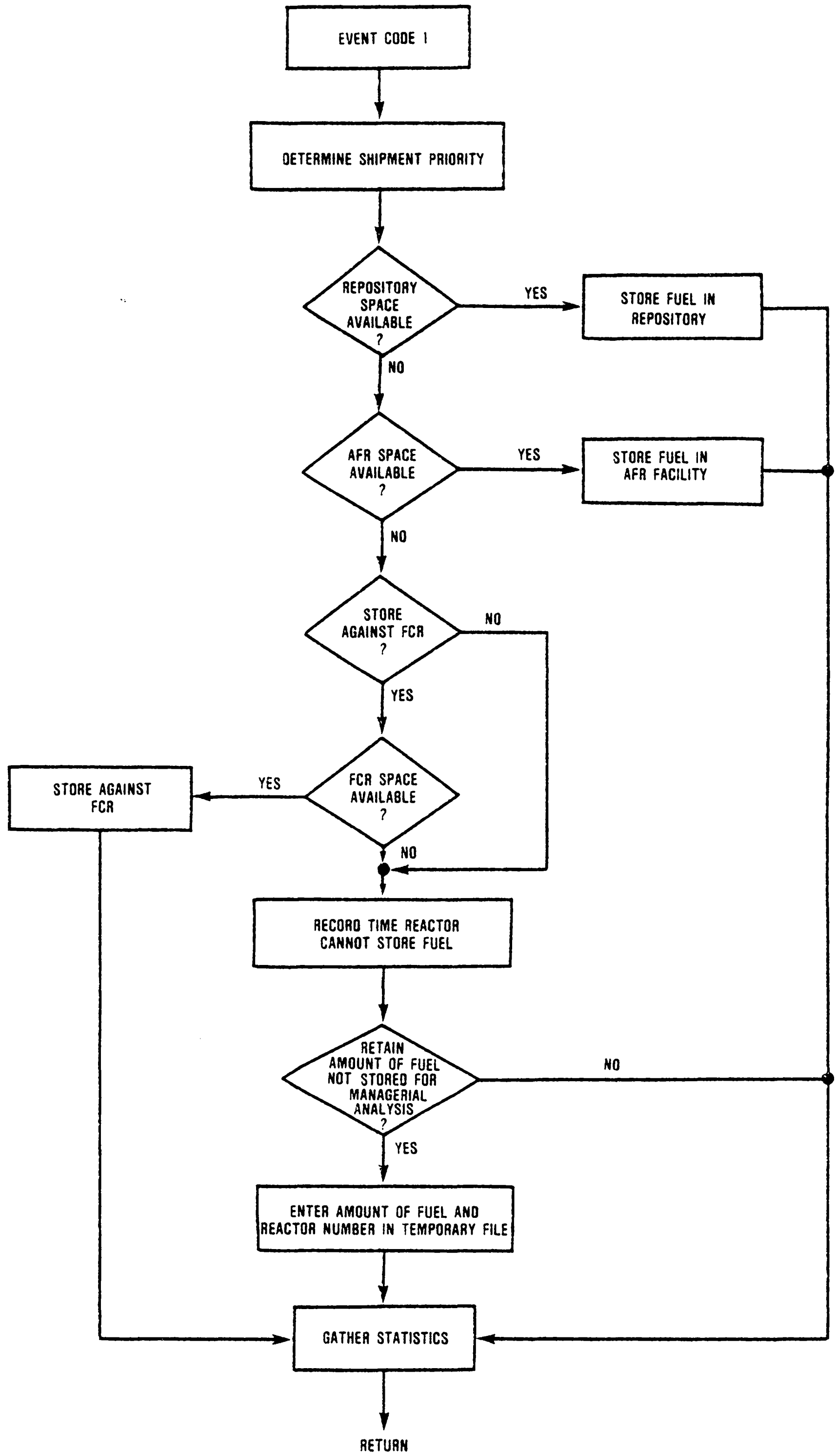


Figure 13. Storage Demand Module of System Two

Unavailable indicates that either the capacity of the storage location is full or that the yearly handling rate of the location is exceeded.

Each reactor and each storage location has its associated longitude and latitude stored in separate arrays. When an event code of one is detected, the distance between the specific reactor and each AFR facility and each repository is determined. Then a sorting routine is used which results in two arrays. The first array contains, in order of nearness, the AFR facility codes and the second array contains the repository codes, also in order of nearness. In an analogous fashion, the AFR facilities and repositories are ordered by current utilization. The utilization is ordered from lowest to highest.

Shipment option five provides the user with extreme flexibility in directing spent fuel shipments. Each individual spent fuel discharge is categorized in four ways: 1) by reactor identification number, 2) by utility identification number, 3) by state code, and 4) by National Electric Reliability Council (NERC) code. The best way to explain this option is by way of several examples. The first example states that discharges from reactors in Virginia, South Carolina, and New York be stored at the Barnwell AFR facility if available storage space exists. If storage space is unavailable, then ship to the nearest facility with available space. All other reactors are to ship their spent fuel to the AFR facility with the lowest utilization. A second example combines several codes. Reactor discharges from the utilities Boston Edison, Baltimore Gas and Electric, and New England Power ship to the AFR facility at West Valley, New York. Reactor discharges from the states of California, Nebraska, Ohio, Washington, and Indiana ship to the AFR facility GE-Morris. Reactor discharges from the reactors Oconee, River Bend, Vogtle, Turkey Point, Beaver Valley, and Cherokee ship to Repository Number Two. Any spent fuel discharge not specifically allocated is sent to the nearest AFR facility. If the nearest facility is unavailable, ship the spent fuel to the facility with the lowest utilization. In addition, any spent fuel discharge not specifically allocated is transported to the repository possessing the lowest utilization; and if unavailable, the next lowest utilization. The user is cautioned to avoid conflicts; that is, designating San Onofre to ship to GE-Morris and all reactors in California to send fuel to West Valley causes difficulties.

During the remainder of the description on storage demand, a shipment priority of nearness is assumed. The methodology for any other shipment mode is identical. Referring to Figure 13, the present policy of the DOE is, if possible, store fuel first in a repository. Since the expected earliest date for the operation of the first repository is 1997, numerous discharges will be shipped to AFR facilities prior to 1997. This expectation is obvious, but crucial, for the existence of an expectation of discharges before 1997 prompts the need for the entire AFR Program. The nearest repository is examined, and if space is available, the discharged spent fuel is stored in the repository. Before returning simulation control back to the GASP IV subroutines which search for the next entry, statistics are collected. These statistics are reviewed at the end of this section after all paths of the storage demand flow chart are travelled. If the nearest repository is unavailable for storage, the next nearest repository is examined and so on. If no repository is available to accept the discharge, the simulation examines the AFR facilities.

In a completely identical manner, each AFR facility is scrutinized for availability of storage space in order of nearest, next nearest, and so forth. If a facility is able to store fuel, the spent fuel is indeed stored and the proper statistics collected. Whenever there does not exist any AFR facility for storage, the full core reserve (FCR) option is simulated, if selected by the user. The FCR option first checks the amount of spent fuel stored against the FCR of the reactor in question. If enough storage space is present, the spent fuel is stored against the FCR capacity and statistics are collected. If the total FCR space or the available portion of the FCR is not sufficient to store the discharge, the next step is to determine if the temporary file option has been selected by the user. Storing spent fuel in this temporary file is a managerial tool. Remember, to reach this point there are no repositories or AFR facilities available and, if employing the FCR option, no FCR storage space. At this point in time the objective of the AFR Program cannot be met since all realistic avenues of storage prove futile. The overriding function of the simulation model is to provide the DOE managers with a mechanism to examine this failure point under the storage scenario being reviewed. Depending upon the magnitude of the failure, the manager can change a voluminous number of factors such as shipping priorities and/or on-line times for AFR facilities in order to eliminate the failure. The temporary file provides a record on how bad the failure is. In other words, a case where only one metric ton of spent fuel cannot be stored is far different than a case where one thousand metric tons of spent fuel cannot be stored. The temporary file option tracks all spent fuel which cannot be stored. If this option is not utilized, the discharge is not recorded, no statistics are collected, and the next entry in the event file is sought.

A variety of statistics are collected after the spent fuel discharge is acknowledged. Prime among the statistics is the percent utilization of the AFR facility or repository. If a facility possesses a low utilization, then the need for its total capacity may not exist. Any capacity to store spent nuclear fuel is rather expensive, and capacity not utilized is wasted. This is especially applicable to AFR facilities. AFR facilities are only interim resting places for spent fuel. After the repositories bear the burden of storage and the spent fuel within the AFR facilities is moved to the repositories, the AFR facilities will be decommissioned. Excess capacity which is provided at a high cost, but not utilized, represents inefficient management techniques. Therefore to examine this important variable, yearly printouts of the percent utilization for each AFR facility and for each repository are provided along with graphs showing supply and demand versus time over the life of the simulation. Examples of these outputs are given in Chapter Five.

Other statistics collected at the end of the storage demand module include percent utilization of FCR, number of shipments to each AFR facility and each repository, and the amount of fuel handled against the yearly handling rates. In addition, the first time spent fuel cannot be stored is recorded and this value is marked on the graphical output. All statistics are printed out either during the simulation period or at the end of the simulation. After the statistics are gathered, the storage demand module transfers control back to GASP IV. A measurement is performed to check end-of-simulation. If the simulation is not over, the next entry in the event file is extracted and the appropriate action undertaken.

The correctness of the storage demand module is ensured due to exhaustive testing performed during model development. The subroutine to calculate a distance between a reactor and facility or repository is verified by hand. The distance is determined by using Great Circle calculations. Numerous examples from other sources provided sample calculations with which to compare the results of the subroutine.[30] Sample supply and demand figures verify the percent utilization routines. The sorting routines are confirmed by temporary write statements within the model. During a simulation run, each time the sort routine is engaged the reactor under consideration and the sorted arrays are printed. Hand calculations confirm proper sort. All other subroutines are similarly checked and verified by hand calculation. After the entire module is unitized, once again hand calculations verify that the sum is indeed equal to its parts. A sampling of data is generally used for verification analysis since the number of discharges in the sample prove manageable.

The next event code to be enlightened upon is storage supply, the topic of the next section.

Storage Supply Module

This section describes the flow of happenings whenever a storage supply event occurs. This storage supply includes either an AFR facility storage supply or a repository storage supply. The discussion assumes an AFR facility supply; however, a completely analogous flow transpires if a repository supply occurs. An AFR supply is signified by an event code of two and a repository storage supply is indicated by an event code of five. Different event codes for a facility supply or a repository supply are used to direct the program to different formatted output statements. Figure 14 details the flow diagram for the storage supply module.

An event code of two indicates that an AFR facility is 1) coming on-line, 2) increasing its existing storage capacity, and/or 3) changing its spent fuel handling rate. Each facility possesses a unique numerical code which is stored as the forth attribute of the entry. The first step in a storage supply is to determine which facility is being modified. Next, the capacity is increased and the handling rate adjusted. Both actions are always performed; however, if only the handling rate is desired to be changed then the user codes the change in capacity as zero. This concept applies especially to repositories, since the initial handling rate is expected to be 1800 mtu per year. Five years after the repository begins operation, the handling rate is expected to increase to 6000 mtu per year. The time, capacity, and handling rate are all automatically extracted from the AFR Program MIS by the use of a series of auxiliary programs. This step is performed during the operation of System One, as

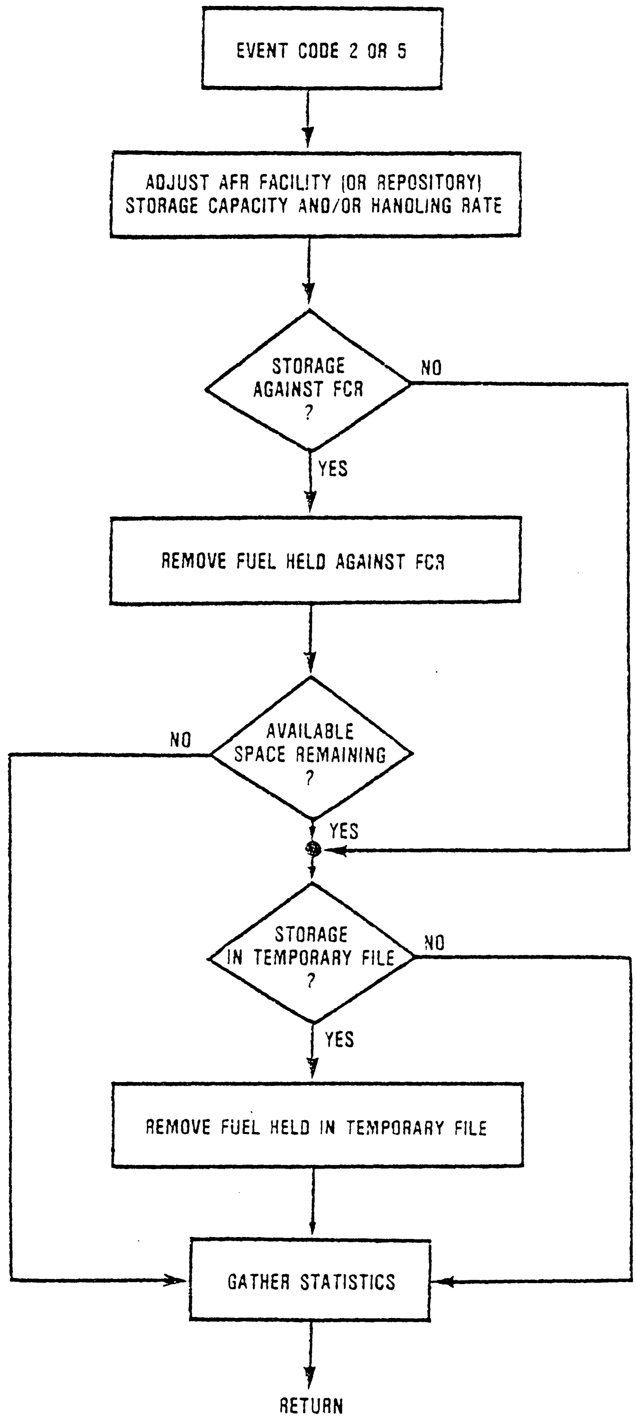


Figure 14. Storage Supply Module of System Two

explained previously. Now if both the FCR option and the temporary storage option are not requested, statistics on facility utilization are collected. After the statistics are gathered, GASP IV regains control and searches for the next entry.

When the FCR storage option is implemented and after the modifications to the facility are performed, an investigation is made to determine if any reactors are storing fuel against their FCR capacity. If this is the case, then spent fuel is removed from the FCR space and stored in the AFR facility. The priority for removal is: the reactor with the highest percentage of its FCR being utilized removes its spent fuel first, regaining FCR capability. Some concern exists that the priority should be the reverse, which in turn leads to a greater number of reactors regaining their FCR capabilities. The reasoning dictating the highest percent utilized order is one of economics. The probability of any individual reactor having to immediately utilize its FCR storage space is independent of any amount of fuel being stored against the FCR. Therefore, if such an accident occurs which requires core removal, the reactor with the greatest utilization has to ship the most fuel to another at-reactor (AR) or AFR storage location. This shipment translates into increased consumer cost and increased down-time.

Removing fuel from FCR storage space continues until either the AFR facility's capacity is exceeded, the AFR facility's yearly handling rate is exceeded, or all reactors have regained FCR storage space. Under the first two instances the remaining action is to collect statistics and search for the next entry. However, if AFR storage is still available and the user selected to track non-storable fuel in a temporary file, then another routine is entered. This routine removes fuel from the temporary file and stores said fuel in the AFR facility. Spent fuel is tracked by reactor within this temporary file, and removed by reactor also. Removal from the temporary file continues until the capacity of the AFR facility is exceeded, the handling rate of the facility is exceeded, or the temporary file exhausted.

As previously explained, the temporary file option provides a mechanism to track the spent fuel which cannot be stored in an AFR faciltiy, a repository, or the FCR portion of the AR storage pool. The user can functionally interpret the fuel contained in this file in any manner so desired. The three options of storage (AFR, repository, and FCR storage) are either being presently employed (FCR option) or will be utilized in future endeavors. Other methods of interim storage are not yet clearly defined and the purpose of the temporary file is to simulate whatever additional methods of storage the user requires. One possible function is transshipment of fuel, as described in Chapter Two. At present, specific guidelines on the methods of transshipment do not exist. An example of such a guideline is a series of rules detailing which reactors transship to which other reactors' AR storage pools. Without such guidelines, programming to account for transshipment cannot be performed. When using the temporary file to simulate spent fuel storage, the feature that is lost is what exactly happens to the fuel. The file just contains so may metric tons of uranium of spent fuel discharged from reactor number X. There is no indication as to where this fuel is to be shipped or how the fuel is to be handled.

The option exists that, upon storage supply and after all reactors have regained their FCR capacity, fuel stored in the temporary file can be removed and placed into the AFR facility. When transshipment is considered, this feature simulates a priority with which to remove the transshipped fuel.

Present policy suggests that whenever the temporary file option is employed, transshipment is simulated.[31]

Spent fuel is shipped to another reactor's AR storage pool if space is available. This space excludes the FCR capacity. Hence, transshipped fuel does not infringe upon FCR capabilities. This policy determines the order in which spent fuel is removed from the FCR areas and the temporary file. Whenever an AFR facility comes on-line or increases either capacity or handling rate, fuel stored against FCR is shipped to the facility. Then if the AFR facility still has available capacity, spent fuel accounted for in the temporary file is removed and stored at the AFR site.

Returning to the storage supply flow diagram; after the temporary file is emptied, statistics are collected. Percent utilization and number of shipments are determined and collected. Fuel stored against the yearly handling rates is accumulated for future reference. Finally, control of the model is returned to GASP IV which locates the next entry in the event file.

This explanation of the storage supply module is presented using an AFR facility as the source of supply. The procedure is identical when a repository comes on-line, changes its handling rate, or increases its capacity.

The accuracy of this module is checked in a similar fashion, as previously described in the review of the storage demand module. Each subroutine is individually examined and verified. Finally, the entire module is checked via the use of sample data and hand calculations.

Utilization Reporting Module

An event code of three directs the program to calculate and report on the percent utilization of each AFR facility and repository. This event occurs once each year. The percent utilization is determined every time a storage location accepts spent fuel or changes its characteristics. This value is equal to the current amount of stored spent fuel divided by the current capacity of the storage location. This value is passed to a GASP IV supplied subroutine. Then at any time requested by the user, GASP IV reports the mean, standard deviation, and maximum and minimum of the sampled values.

The basic assumption in these calculations is that the variable (in this case, percent utilization) has a constant value during the interval from one report of utilization to the next time a value is reported. Hence, the relative frequency with which the variable has a specific value can be considered as the proportion of the time during the simulation that the variable has that value.[21] This translates into the following equation

Average Value = sum over x[x*t(x)]/TOT

where x is the value of the variable; t(x) is the time during the simulation that the variable has the value of x; and TOT is the total simulation time.[21]

This equation is modified due to the value of TOT. GASP IV defines TOT as the total simulation time; however, if an AFR facility or a repository does not come on-line at the beginning of the simulation (usually 1978), then erroneous results follow. For example, if a repository comes on-line in 1997, GASP IV assumes the value of x as zero from 1978 until 1997; when, in fact, the value of x is non-existent until 1997. To overcome this difficulty, a separate array is formed which contains the on-line times of each storage location. Then whenever the utilization values are reported, instead of TOT representing total simulation time, TOT is directed to indicate simulation time from the appropriate storage location on-line time to the present time.

Hand calculations confirm that the modification works as anticipated. Examples of utilization reported output and a discussion on how these values can benefit the DOE managers are given in Chapter Five.

Plot Data Module

GASP IV provides for up to ten line printer plots. Each plot consists of one independent variable and up to ten dependent variables. In addition, all plot data can be tabulated during final output. In this model, the independent variable is time and the dependent variables are the supply of and demand for spent fuel storage of each individual storage location. An event code of four flags the collection of data. This event is set to occur every two-tenths of a year.

Line printer plots are very quick and easy to produce. In addition, a standardized language can have the plots produced at most any computer faciltiy. However, plots made by a printer are usually difficult to read and understand. In order to illustrate better the supply and demand, use is made of an electrostatic plotter. The actual GASP IV plots are suppressed, but the tables of plot data are printed. An auxiliary program automatically extracts the data from the tables and produces a series of plots generated by the electrostatic plotter at Virginia Tech. This series includes a plot for each AFR facility, for each repository, for the total AFR supply and demand, and for the total repository supply and demand. Figure 15 is a sample line printer plot and Figure 16 is a plot produced on the electrostatic plotter. Both plots originate from the same data, yet the electrostatically drawn figure is noticeably cleaner and easier to interpret.

AFR Spent Fuel Removal Module

The final event code to be covered is number six. Remember that the numerical value of any event code has no relation as to when that code is invoked. Event code four indicates plot data collection and occurs every two-tenths of a year. Event code two, implying AFR facility supply, happens only when an AFR facility undergoes a change in status. Returning to code six, this produces the action of transferring spent fuel from an AFR facility to a repository.

The AFR facilities are interim storage areas which

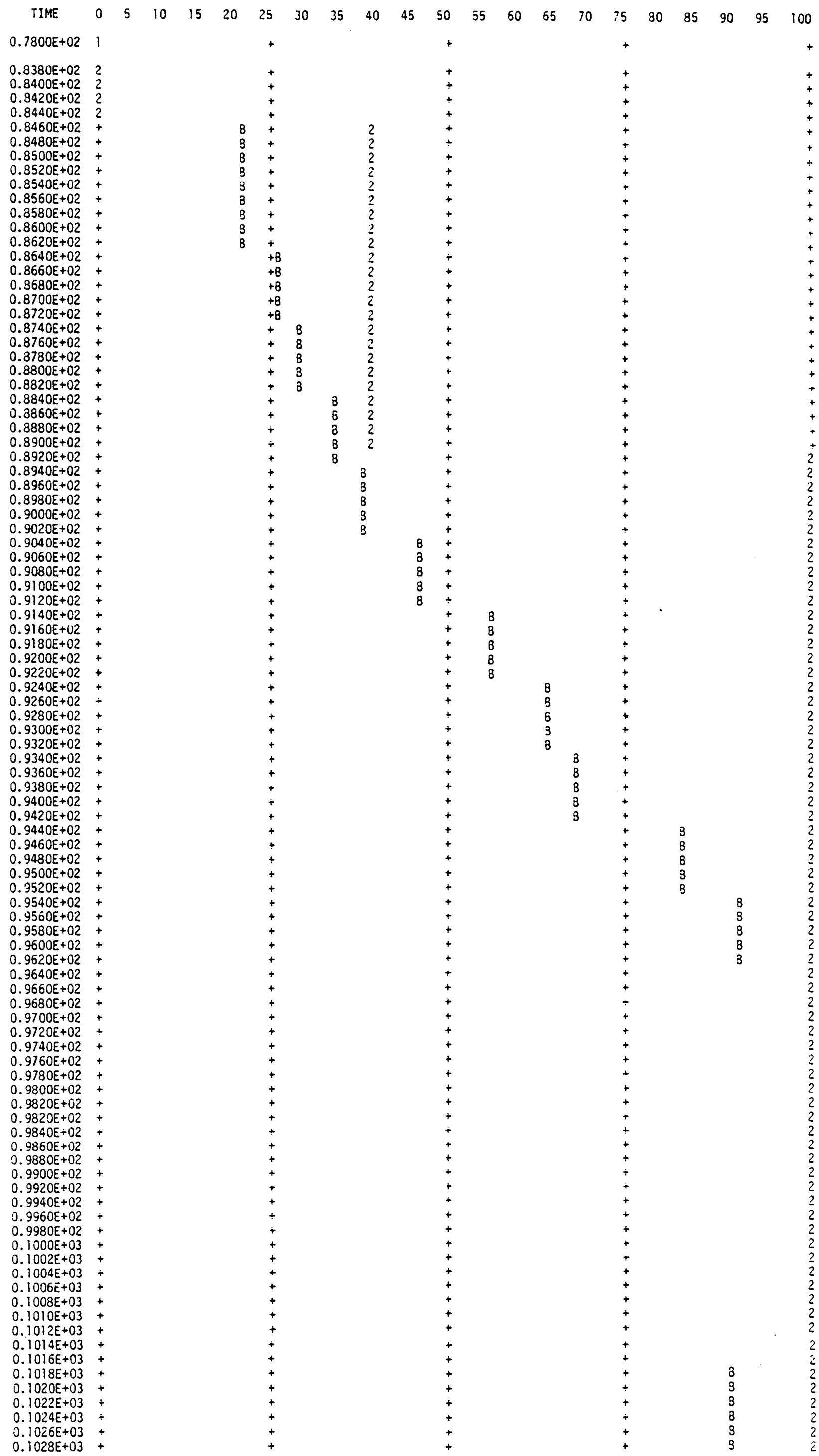


Figure 15. Storage Utilization for the Morris Facility

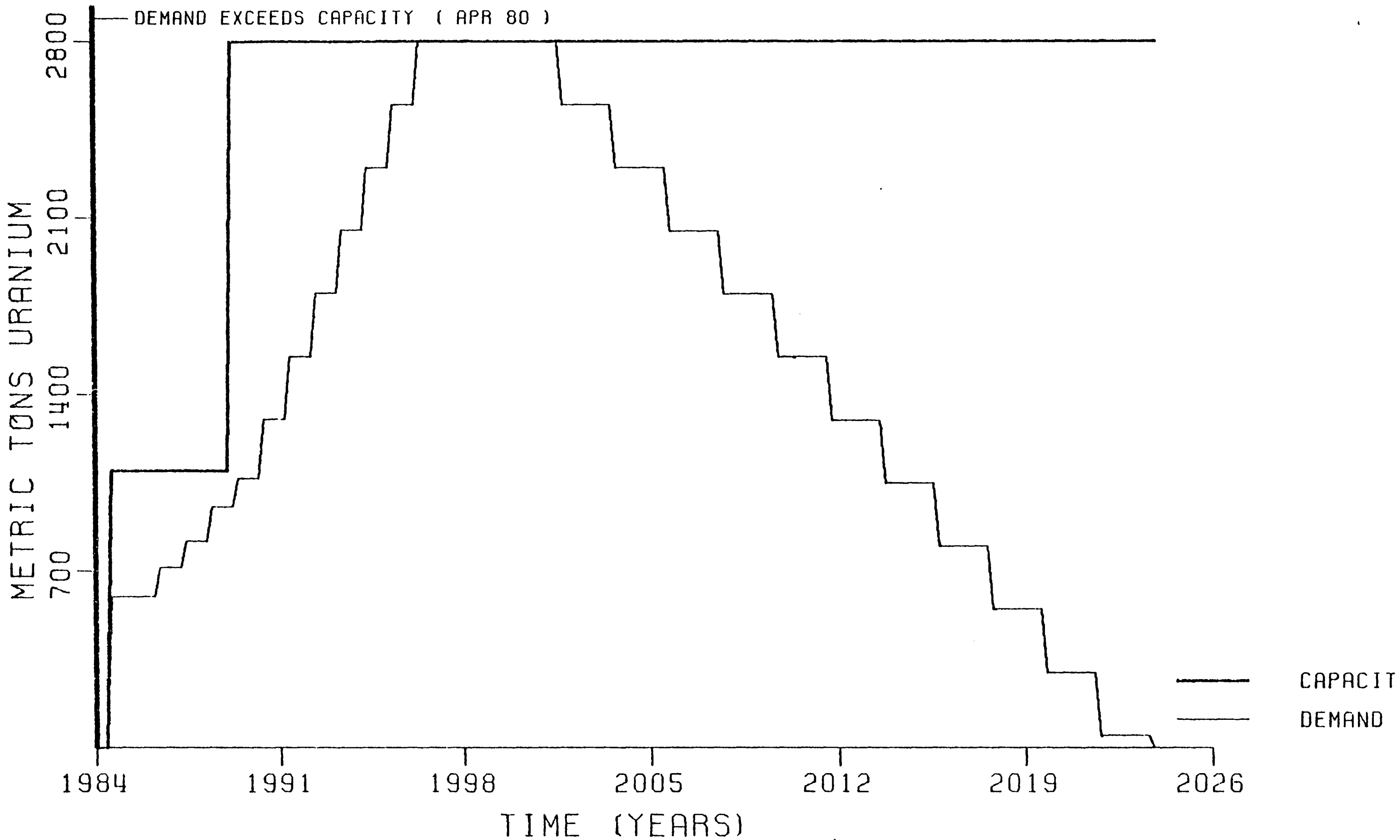


Figure 16. Storage Utilization for the Morris Facility

are to be decommissioned after the spent fuel is transferred from the AFR to the geologic repositories. Upon examination of the timetable for repository operation, this decommissiong happens well into the next century. The handling rates of the repositories are based on projected amounts of waste material from individual reactors and defense programs. Remaining repository capacity is utilized for the transfer of fuel from the AFR sites to the repositories. The simulation of this transfer is offered as an option.

Figure 10 illustrates the flow involved in simulating the transfer of spent fuel from the AFR facilities to the repositories. The first action is to check the status of the repositories. If a repository has available space (i.e., neither the total capacity or yearly handling rate are exceeded) then AFR stored spent fuel is removed from the AFR site and stored in the repository. This process continues until each AFR facility and each repository are canvassed.

Shipments from a specific AFR facility to a repository can be halted if 1) the capacity of the repository is exceeded, 2) the yearly handling rate of the repository is exceeded, 3) the yearly handling rate of the AFR facility is exceeded, or 4) the AFR facility ships all its stored spent fuel. In prioritizing the AFR facilities, the facility with the highest percentage of utilization is the first to ship to the available repository. This priority is selected since the facilities with the highest utilization probably come on-line at the earliest times. Hence, the decommissioning of the older AFR facilities occurs first. After each available repository receives fuel from the AFR facilities, the usual statistics are collected and control is relinquished to GASP IV.

Once again verification of this module is performed through the use of temporary write statements, abbreviated data, and hand calculations.

All possible event codes have been examined. In each case, after the appropriate action is accomplished, control is turned over to GASP IV. If the simulation is to continue, GASP IV checks the event file for the next entry. There are several methods to terminate the simulation model, all explained in the next section.

<u>End-of-Simulation Module</u>

There exists three normal methods and one abnormal wrinkle in which to halt the simulation model. The three normal methods are

1) setting a GASP IV variable to minus one,
2) defining a time (e.g., 1995) to halt the simulation, or
3) emptying all entries from the event file.

The abnormal wrinkle involves any method causing a termination error. This method is not as foolish as said method may at first appear. This is especially useful during certain testing since causing an early termination via an illegal statement may be the easiest technique. One example is to limit the time of simulation. Here time refers to computer execution time. If one is interested in testing just the beginning of a routine, the execution time can be set at a low value. For normal operation, an increased time limit must be coded.

For the fuel cycle simulation model, the termination of the simulation occurs at the user inputted end date. The event file never becomes completely empty since certain events are automatically entered on a regular basis. This logic applies to event codes three and four. Considering code three, after statistics are reported, the current time is increased by one; and this new time, along with an event code of three, is entered into the event file. This relieves the user of having to input an event code of three each year throughout the desired simulation period. Therefore, entries of discharge data may expire, but there still exist some events due to the automatic replenishing feature. The user is cautioned not to rely on the model ending when entries no longer exist.

If the simulation is not over, the next entry is extracted from the event file and the flow of Figure 10 repeated. Whenever the simulation is terminated, several output routines are executed. GASP IV first looks for a user supplied output subroutine. This subroutine is optional. The fuel cycle model employs such a routine to report on the number of shipments to each AFR facility and each repository. Also included in the output is an echo check of the input data from User Input. This feature aids the user in keeping track of the output from several different simulation runs.

After the user supplied output subroutine completes its deeds, the GASP IV provided subroutine is called to duty. A detailed listing of the options available with this routine is given in the GASP IV reference. For this model, the main use for this subroutine is in producing the plot data in tabular form.

This completes the description of the simulation model. The next section examines model verification, as pertains to the entire model. The output, along with numerous examples, is tackled in Chapter Five.

MODEL VERIFICATION

During the discussion of each module comprising the entire simulation model, a paragraph is devoted to verification of that module. After all modules are linked together, model verification proceeds in a similar fashion. The printed output is scanned for any FORTRAN execution errors. Once all such errors are corrected, detailed hand calculations are performed. Printouts of all the input are amassed and manual walk-throughs undertaken to simulate the simulation model. The errors in coding and logic are discovered using this methodology. Corrections are made and the final model certified. Appendix B details this procedure by providing a sample input and the corresponding output along with a written explanation.

CHAPTER FOUR - INPUT VERIFICATION

Employing a snapshot of data for use as input has always been shunned upon; for this input would quickly become obsolete. The activities and their interrelations within the AFR MIS can change monthly; estimates on foreign discharge requirements are subject to political decisions; and the repository data can become more detailed and complete. To insure that the user of this simulation model is employing the most current data, specific mechanisms are developed and implemented. The mechanisms fall into two general categories: 1) those methods which pinpoint gaps and/or overlaps in interrelationships and 2) those methods which yield an easily updatable base of information. These mechanisms are described, where applicable, in relation to the AFR MIS, the reactor discharge data, and the repository data.

Some general comments deserve mentioning. First, the following discussions, centering upon the various sources of input, are concerned only with those aspects relevant to the simulation model. The purpose of this paper is to describe the model and not a general explanation of the AFR MIS or the DISFUL program. The reader is encouraged to obtain the referenced works whenever more information on these subjects is desired. Second, the author has served as technical manager for the development of the octopodous AFR MIS and all associated tentacles. Since numerous computer programs which scrutinize the MIS have multiple applications, the actual FORTRAN coding of some of the programs has been completed by other individuals. However, all the program designs have been developed and supervised by the author.

AFR MIS INPUT

In October 1977, the Department of Energy (DOE) announced that under certain conditions the federal government would take title to and store spent nuclear fuel from private commercial reactors.[1] All costs associated with the storage, including AFR decommissioning costs, are borne by the utilities concerned. This effort requires a computerized MIS in order to track and report on the large number of work packages (e.g., contracts) required to successfully meet the AFR Program objective. Each work package allocates to a specific contractor a portion of the total effort. Areas of effort include 1) legislative requirements; 2) site-selection studies; 3) transportation evaluations; and 4) facility design, construction, and operation requirements. Presently, the AFR MIS contains information on approximately ninety such work packages. The effort allocated to each work package is further broken down into specific elements called activities. A work package contains, on the average, twenty activities. Through predecessor and successor relationships, the activities are related not only to the activities within their own work package, but to any work package or packages within the AFR Program. For example, a specific licensing activity in a work package within the general area of legislation has as a successor an activity contained within a work package under facility design. These relationships form the links which permit the tracking of the activities contained within the AFR MIS, culminating in the AFR facility operation activities. Hence, a delay of six months in site selection can be translated into the delay for the on-line time of a new AFR facility. Or, a slippage in reracking design and/or licensing can have a domino effect through the MIS affecting the anticipated capacity of the GE-Morris facility. The correct identification of all related activities is of prime importance to insure the accuracy of the output from the simulation model.

The orginial framework for the AFR MIS is based upon a functional work breakdown structure (WBS). This structure places work packages and their associated activities into functional categories. A sample of this functional structure is shown in Figure 17. This functional WBS is employed in the allocation of funds through contractual processes. Note that the overhead, Planning and Management (task 11), and the Internal Research and Development (IR&D), Technical Support (task 14), are of such importance to long-term R&D programs that there is both political and managerial value in highlighting them in this structure. However, these tasks are merely auxiliary functions in support of the overall AFR Program objective. Typically, decision points and integration milestones are placed under Planning and Management; scoping and evaluation milestones, under Studies and Analysis; and concept development milestones, under Technical Support. In addition, milestones related to the objectives and products of the program are intermixed throughout the functional WBS. Although this functional WBS satisfies the financial allocation requirements of the AFR Program, program control problems can arise from this functional dissemination of

11 Planning and Management

- **111 Planning**
  - 1111. Facility Planning
  - 1112. Program/Budget Planning & Management
  - 1113. Business Services
- **112 Planning Support**
  - 1121. Full Core Reserve Evaluation
  - 1122. Storage Logistics
  - 1123. AFR/AR Economics
  - •
  - •
  - •
- **113 Fee**
  - 1131. Fee Methodology
  - 1132. Fee Update
  - •
  - •
- 114 Storage Contract
- 115 Program Management
- 116 Quality Assurance
- 117 Public & Governmental Relations

12 Facility Acquisition

- 121 Morris
- 122 Barnwell
- 123 West Valley
- 124 New AFR #1
- 126 New AFR #2

13 Legislative Requirements

- 131 NEPA Activities
- 132 Congressional Activities

14 Technical Support

- 141 Safety & Environmental Technology
- 142 Dissassembly & Fuel Rod Storage
- 143 Equipment Development
- 144 Storage Alternatives
- 145 Operations Data & Assessments
- 146 Generic Studies
- 147 Safeguards

15 Transportation

- 151 Transport Industry Programs
- 152 Adequate Transport Capability
- 153 Transportation Support

16 International Activities

- 161 Transfer Plans
- 162 Multinational Storage Facility
- 163 International Cooperation
- 164 Activities Integration & Planning

Figure 17. Functional WBS of the AFR Program

milestones. In fact, technological, operational, and managerial gaps and overlaps can exist whenever a functional WBS is solely employed in program management. During the design phase of the tracking and milestone extraction module, the elimination of these gaps and overlaps becomes necessary for proper module operation. To assist in this elimination, a modified objective WBS and a logic flow diagram are program customized.

A classical objective WBS is not set against a time base. However, to aid in the tracking and aggregation of milestones a time base is desired. A modified objective WBS is developed during the design stages of the simulation model in order to visualize the timing of the individual activities within the AFR MIS. Figure 18 illustrates an abbreviated section of the AFR Program modified objective WBS. The top portion (above the dotted line) is a typical objective orientated structure. This portion is linked to the individual activities which are placed against the vertical time grid. The upper portion provides the objective structure, crucial in the process of checking gaps and overlaps. The lower portion is invaluable in providing the mechanism with which to track program progress via the simulation model. In addition, slack times can be observed and considered in the

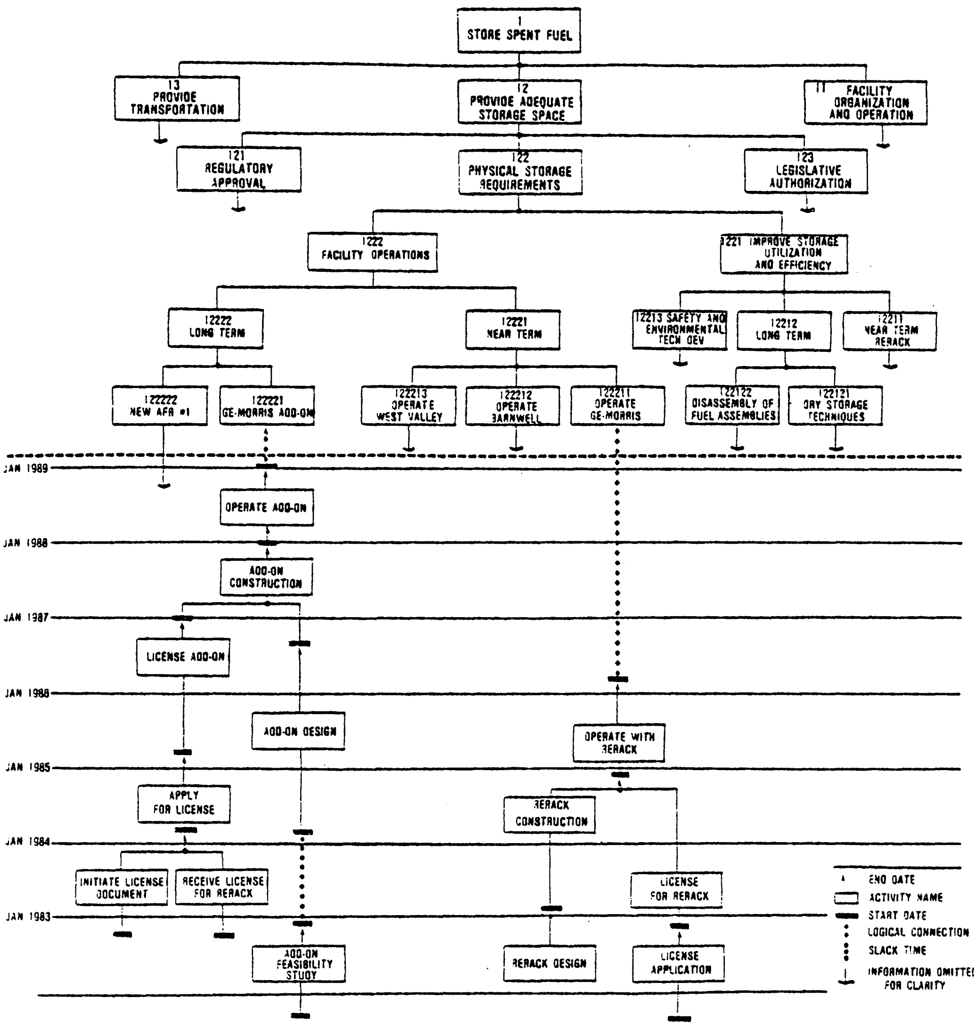


Figure 18. Modified Objective WBS of the AFR Program

managerial processes.

One feature lacking in any objective diagram is an ability to handle decision branches. Decision branches highlight gaps which may not be apparent in the objective WBS. A logic diagram provides a branching mechanism, as shown in Figure 19. Whereas the WBS's are based on success-oriented management, the logic flow diagram allows for alternative directions and result decisions.

The combinatorial use of these three WBS's and the logic flow diagram ensure that the financial aspects (and hence, resource allocations) and all the decision points, the milestones, and the interrelationships of the AFR Program are realized, maintained current, and trackable. The importance of maintaining current data and ensuring no gaps or overlaps exist can not be understated since these milestones become the decision variables of the GASP IV simulation model.

The AFR MIS is a continuously evolving system. Work packages are completed; new work packages are awarded; and activities can be redirected, reduced, expanded, or eliminated. Remember, the managers can employ any available resource to alter the status of any activity under their jurisdiction. Hence, the objective structure is constantly employed to guarantee a unitized MIS. As

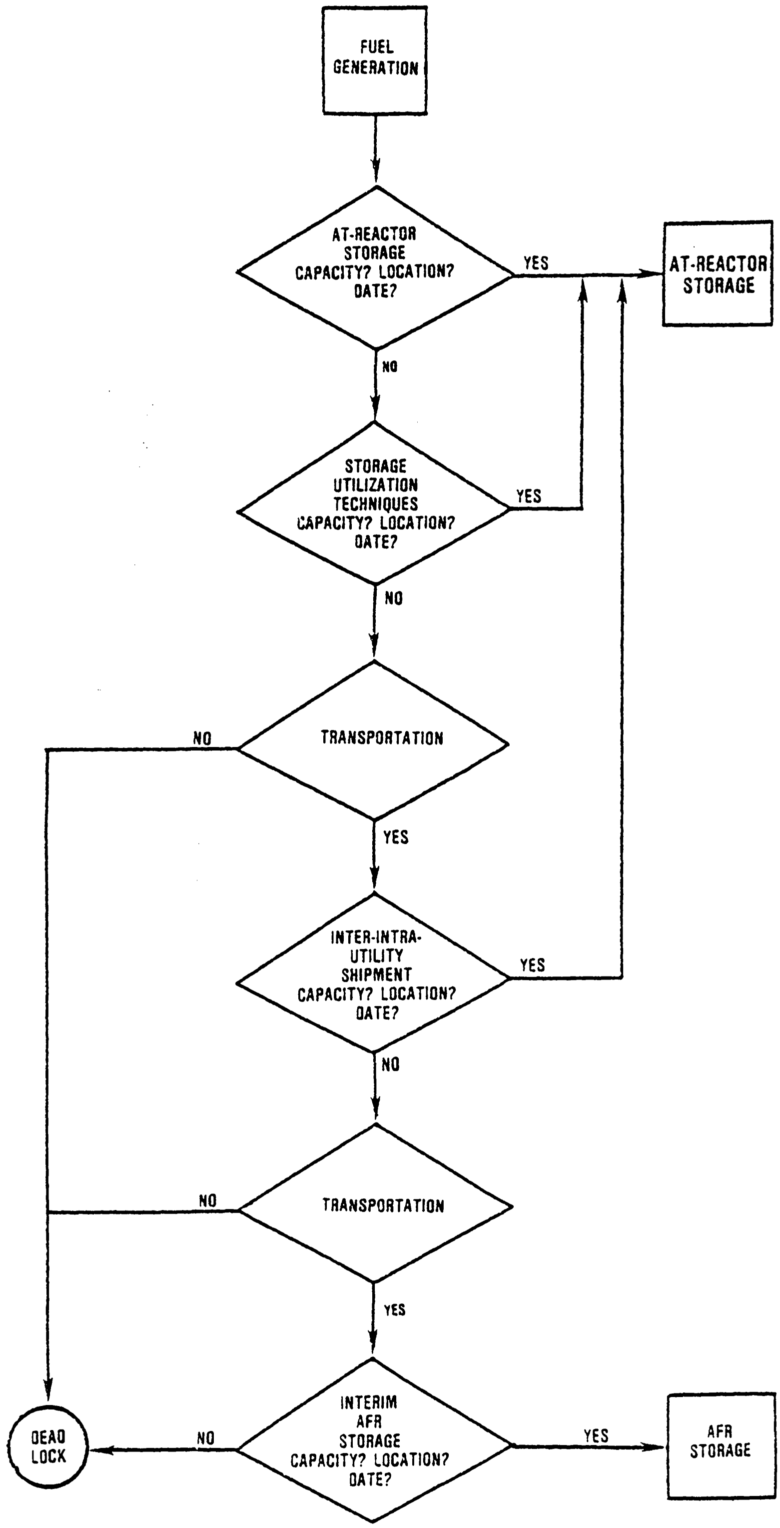


Figure 19. Logic Diagram of the AFR Program

explained in Chapter Three, the tracking module of the simulation model works within the MIS. So as the system grows, the simulation model always uses the best available information.

Due to the scope of the AFR Program, the work packages are awarded to numerous contractors. Each contractor is responsible for the exactness of the activities under his control. To ensure that the MIS possesses current data, all contractors must report their work progress on a monthly basis. For even when all present activities are structured correctly, if the information on a work package is outdated, the output of the model becomes unreliable. Several techniques are designed and implemented to relay the information contained within the MIS to the contractors. In turn, the contractor can verify said data. At the onset, the concise presentation of a contractor's information is of paramount importance. Interviews with different contractors reveal that they do not always have enough time to review lengthy monthly reports. Two visualizations of activity data are developed to meet the requirement of conciseness.

The first visualization is a totally computerized version of the DOE form 535, as shown in Figure 20. This

FORM DOE 535
(1/1/78)
VPI FACSIMILE

U. S. DEPARTMENT OF ENERGY
SAVANNAH RIVER OPERATIONS OFFICE
MILESTONE SCHEDULE AND STATUS REPORT

PAGE 1 OF 1

1. CONTRACT IDENTIFICATION STORAGE LOGISTICS	2. REPORTING PERIOD 01 APR 80 THROUGH 30 APR 80	3. CONTRACT NUMBER 1122 .
4. CONTRACTOR NAME, ADDRESS) S. M. STOLLER CORP. NEW YORK NEW YORK		5. CONTRACT START DATE 1 JUNE 78 6. CONT. COMPLETION DATE INDEFINITE

7. IDENT. NUMBER	8. REPORTING CATEGORY	9. FISCAL YEARS: FY 1979 (1 2 3 4); FY 1980 (O N D J F M A M J J A S); FY 1981 (1 2 3 4); 82; 83; 84	10. PERCENT COMPLETE
1	PROVIDE COMP CODE/UP		100
2	PREPARE AND REVISE S		95
3	EXPAND MAINTAIN CO		55
4	REVISE STORAGE LOGIS		0
5	REVISE STORAGE LOGIS		0

11. REMARKS

12. SIGNATURE OF CONTRACTORS PROJECT MANAGER AND DATE

13. SIGNATURE OF THE GOVERNMENT TECHNICAL REPRESENTATIVE AND DATE

Figure 20. DOE form 535

form lists the activities contained within a specific work package. Milestones are shown as triangles or diamonds located upon the activity time bar. Different symbols represent different levels of milestones. This form is forwarded to the appropriate contractor once a month for his review. Upon this form, the person in charge indicates: 1) percent of completion for each activity; 2) whether an activity is behind, on, or ahead of schedule; 3) change in activity start and/or end date; and 4) any remarks. Realize that only outdated information needs to be corrected, freeing the individual of the monthly task of completely filling out from scratch such a form.

The DOE form 535 depicts milestones; however, this form does not illustrate the interrelationships which are vital to the simulation model. A totally original form, termed network diagram, is developed to display such interrelationships against a time line. Figure 21 is a picture of the network diagram corresponding to the work package shown in Figure 20 (DOE form 535). This is a unique form in that the related activities forming the network are shown against a time line. The pentagonal symbols contain the structure numbers of predecessor or successor activities. The elliptical symbols contain the structure numbers of the coordinated activities. Once a

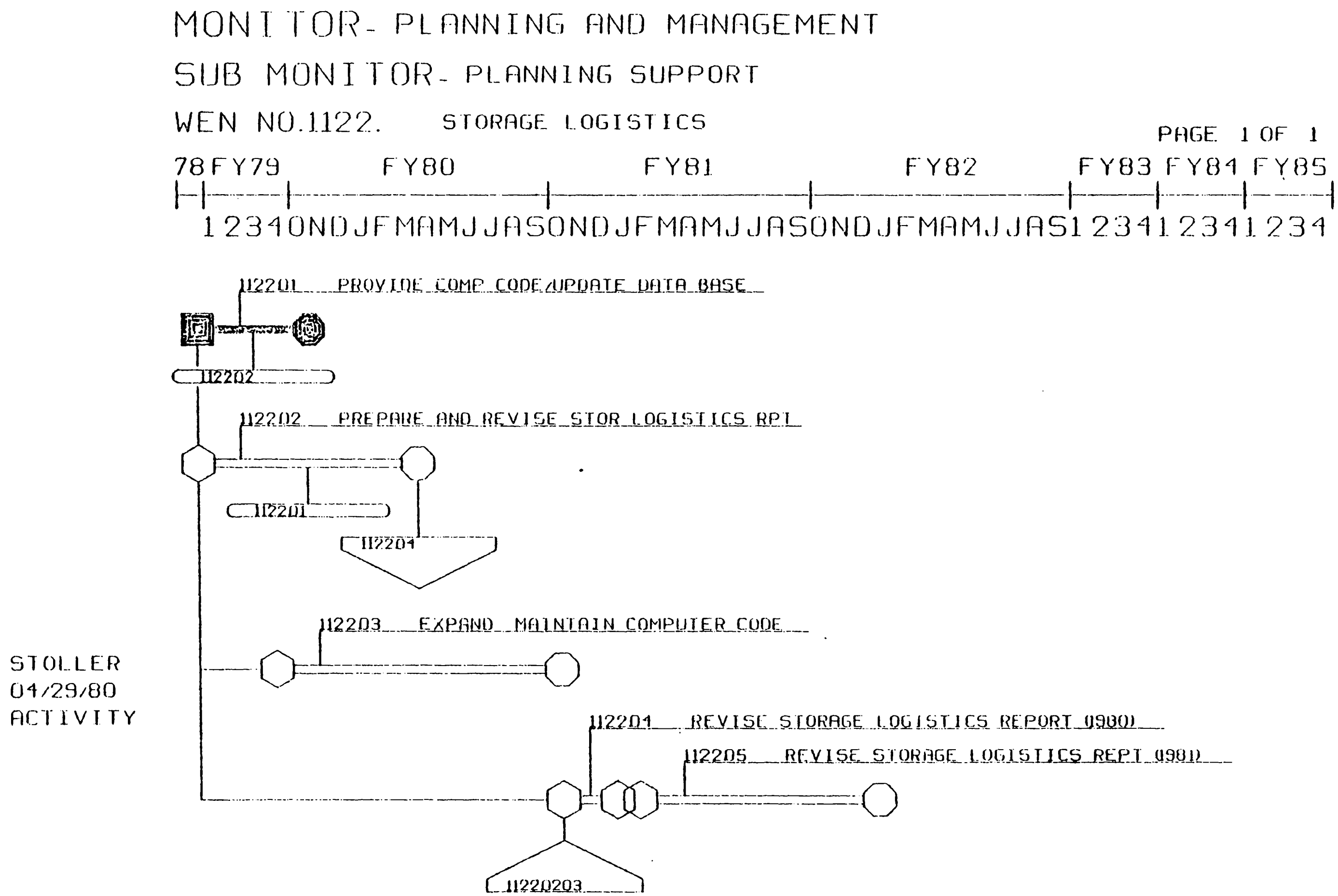


Figure 21. AFR Program Network Diagram

month this form is sent to the appropriate contractor for his perusal. Changes are performed upon the network diagram and then the form is returned to Virginia Tech for inclusion into the MIS.

In summary, an updated AFR MIS is crucial to the success of the simulation model. Programs are designed which track through the activities within the MIS and extract pertinent information required for System Two of the model. Several mechanisms are employed to guarantee the accuracy of the MIS. A collection of WBS's and logic diagrams present the information contained within the MIS from various vantages. This ensures that gaps or duplications are identified and corrected. The DOE form 535 and the network diagram provide the contractors with a mechanism to easily and routinely update the milestones within the MIS. These milestones become the decision variables of System Two. Hence, if the simulation is required six months from now, the user can be assured that the information about the AFR Program has been reviewed by the contractors and approved by the DOE officials within the last thirty days.

REACTOR DISCHARGE INPUT

The domestic reactor spent fuel discharge data originate from the DISFUL program developed by the S. M. Stoller Corporation. The DISFUL program extracts the raw data from the Stoller Database, manipulates the data, and produces the required output. As previously explained, additional information about each reactor discharge is required. Some of this information (i.e., reactor location and full core reserve capacty) is retrieved directly from the Stoller Database and combined with the DISFUL output. As is evident, the Stoller Database is the keystone for the domestic discharge data through the year 1999.

Periodically, a questionnaire is sent, by the DOE, to each utility throughout the United States which operates or intends to operate a nuclear facility. Information on all aspects related to spent fuel discharge is accumulated and the Database updated. Concern is raised by the DOE as to the completeness with which the questionnaires are being returned. This form is multiple pages, and the possibility exists that not every utility has the time to comprehensively review the form. In order to reduce the burden upon the utilities, a questionnaire on each individual reactor is produced and filled-in with data from the Stoller Database. The result is that the appropriate individual has to only correct outdated data, rather than totally complete the form.

Figure 22 is a reproduction of the first page of the questionnaire developed at Virginia Tech. A complete questionnaire can be found in Appendix D. If all the information is accurate, the responsible person initials the form and returns said form to the S. M. Stoller Corporation.

The latest mailing occurred during the summer of 1980. The S. M. Stoller Corporation employed this reporting technique as a prototype. The form which is actually mailed to the utilities is slightly different than that given in Appendix D; however, the philosophy pioneered at Virginia Tech is incorporated into the Stoller questionnaire.

By utilizing this questionnaire, the Stoller Database is periodically updated. Whenever this Database is changed, a copy is immediately forwarded to Virginia Tech. The database at Virginia Tech is then formatted into a MARK IV structure. MARK IV is a generalized report writer, used to extract any datum or combination of data from the Stoller Database.[25] Whereas the DISFUL program

DOMESTIC SPENT FUEL STORAGE SURVEY

DATE: ______

Utility Name: ALABAMA POWER CO
Reactor Name: FARLEY-1
Location (Latitude, Longtitude): 31 Degrees,
85 Degrees, 5 Minutes

Person(s) Supplying Data: ______
Mailing Address: ______

Telephone Number: ______

REACTOR

Type: Pressurized Water XX Boiling Water ______
Vendor: W
Net Capacity: MWe 329. MWt ______
Commercial Operations Date (Month and Year) 1978
If not operational, state:
Licensing Status: ______
Construction Status (Month and Year): ______
. Limited Work Authorization (Construction Permit): ______
Completion (or Forecast): ______
Full Core Size: Number of Assemblies 157
Assembly Weight MTU 0.451

POOL CHARACTERISTICS

Number of Rectangular Spent Fuel Pools: 1

Length (ft) 37.9
Width (ft) 21.6
Depth (ft) 39.0
Center-to-Center Spacing (in.) 10.37500

Figure 22. Domestic Spent Fuel Storage Survey

produces hordes of output desirable for certain applications, the need arises for only specific information. For example, one item of information which may be required is the at-reactor pool dimensions for the Farley-1 reactor. The MARK IV system yields this knowledge quickly and inexpensively. Although MARK IV is primarily employed as described, this system provides an indirect verification of data. The DOE managers possess a detailed understanding of the AFR Program. If an unreasonable value is found on a report produced by MARK IV, then an investigation into the Stoller Database is forthcoming.

By the direct use of periodic questionnaires and the indirect application of the MARK IV produced reports, the Stoller Database is maintained in an accurate state. In turn, the domestic reactor discharge data continuously enjoy a high reliability factor.

The discharge data for the period 2000 to 2020 and the foreign spent fuel data both originate from the DOE documents referenced in Chapter Two. Since the author is the technical manager for the developmemt and maintenance of the AFR MIS, a direct channel of communication exists with the DOE. As the DOE updates either the foreign estimates or outyear discharge figures, these updates are forwarded to Virginia Tech and implemented into the simulation model.

REPOSITORY INPUT

The program to provide comprehensive repository information is in its infancy. The preliminary data which are given in Chapter Two are utilized in the simulation model. As these data are adjusted, the changes will be quickly inputted into the model. This technique continues until a database, similar in concept with the AFR MIS, is developed.

The author has exchanged information concerning MIS's with the ONWI managers who are responsible for the repository program. A similarity between the existing AFR MIS and the proposed repository MIS is apparent. Hence, as the repository MIS forms, the tracking and the milestone extraction programs employed against the AFR MIS, will be applied against the repository MIS. Obviously, changes to the actual FORTRAN programs will be required; however, the logic will be the same for both MIS's.

REFLECTIONS

The primary aim of this work is the production of a simulation model for the back-end of the nuclear fuel cycle. The model provides the device which can be used to forecast and to play what-if games with the nuclear fuel cycle. Whether or not the input data are accurate or not is of little importance to the actual working of the model. If the model performs correctly for the phantasy date of 1979 for the operation of the GE-Morris facility, then the model will perform correctly for the realistic date of 1984. However, the DOE requested such a model and fully intends to exercise same; therefore, from their vantage point, accurate input data is imperative. An analogy would be the design and use of a fuel depletion code utilizing make-believe neutron cross-section data. The code would still perform, but the output would be of little pratical value.

The methods to verify the various sources of input were designed over the past few years; some resulting from work directly related to the simulation model; some, indirectly related. All such methods ensure the existence of a most comprehensive and accurate source of input on the back-end of the nuclear fuel cycle.

CHAPTER FIVE - RESULTS

This chapter is divided into three sections. The first section examines the sensitivity of the model to various permutations. Exemplary output is reviewed in the second section. The third section studies the simulation output obtained when using the best available input as of August, 1980. This section highlights the current status of spent fuel disposition in the United States.

This simulation model is designed as a descriptive model to be used by DOE and/or utility managers as a tool to aid in their managerial responsibilities. Since this descriptive model employs a large number of parameters, there is no intention to exhaust every avenue of variance. Instead, a sampling of parameters which can realistically be varied are selected. The resulting effects upon the model output are then analyzed.

An important delineation is required. This present chapter utilizes two sets of input data. The first set is called sample data and forms the foundation for the simulations which produce the sensitivity analyses and the exemplary output. This output is used to explain how an individual reads and interprets the simulation reports, and not necessarily used to show the current status of the AFR Program.

Although the utilization of the most current data during these analyses is desirable; the effort to produce such analyses requires several months. During this period, some of the input data can be updated. Therefore, the second set of data is amassed just prior to completion of this work. This best-case information yields the status of the AFR Program as of August 1980. The differences between the sample data and the best-case data include shipment priority to the AFR facilities and the number of new AFR facilities. The spent fuel discharge amounts are identical.

Results, as presented in this chapter, are factual accounts of model behavior. Conclusions are interpretations, by the author, based upon the results. Sometimes there exists a gray area between results and interpretations. To reduce any conflicts between factual results and opinionated conclusions, the conclusions are separately presented in the next chapter.

SENSITIVITY

The basic purpose of a sensitivity analysis is to determine those parameters which influence the output and to what extent.[32] Usually, some parameters can vary widely with minimal effect upon the output, while other parameters possess greater sensitiveness. The accident analysis module is looked upon first, with a determination of the relation between the yearly accident rate and the failure point of the AFR Program (i.e., the first time a reactor cannot successfully store spent fuel). Then, model sensitivity to facility handling rate, on-line time, and capacity is analyzed.

Accident Analysis Sensitivity

To analyze the accident analysis module, the yearly accident rate is varied from five percent to ninety percent. At each selected rate, the earliest time at which any reactor cannot store spent fuel is recorded. For each specific accident rate, ten simulation runs are executed in which the seed for the random number generator is changed in order to arrive at an average failure time.

The value of ten runs per accident rate is chosen in regards to computer cost and confidence. An analysis provides an acceptable confidence interval.[33] The ten values of the failure point resulting from the seventy-five percent accident rate yield a ninety-five percent confidence interval of plus or minus 1.7 years. When this confidence interval is balanced against the cost of additional simulation runs, the interval is deemed acceptable.

Figure 23 is a graphical representation of the sensitive relation between yearly accident rate and failure point. Included on the figure is a curve of the equation

$$y=79.32+0.158x-0.00044x^2 \; .$$

This curve is derived by the method of least squares.[34] By applying standard statistical analysis to the values determined by the simulation and the values calculated by the curve, a ninety-five percent confidence interval is determined.[33] The interval varies from minus 1.7 years to plus 1.68 years.

This analysis is performed using the sample input data. The values of the parameters within the accident

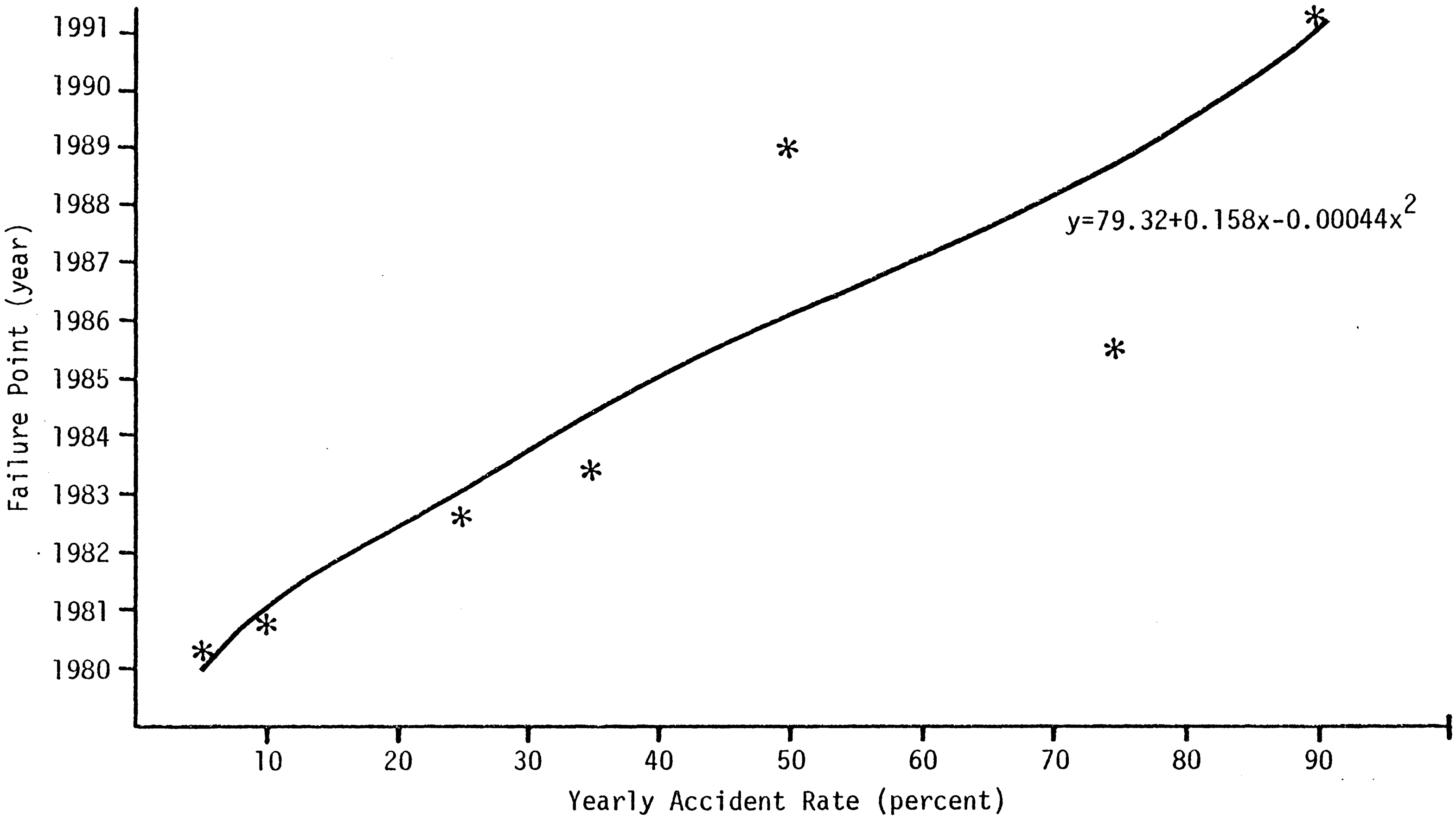


Figure 23. Accident Analysis Module Sensitivity

module are given in Table 8. The coefficients of the various curves are arbitrarily selected to yield a range of values as the independent variables traverse their domains. The accident module adjusts discharges on an individual reactor basis; therefore, a termination time of 99.0 is chosen which corresponds to the last year of individual reactor discharge data.

Facility Parameter Sensitivity

Three parameters which classify an AFR facility are varied and the effects upon the simulation output are measured. The three parameters are facility capacity, on-line time, and yearly handling rate. The measured effects include failure point; percent utilization of the associated facility; and total amount of fuel which cannot be stored in a repository, an AFR facility, or the FCR section of the on-site storage pools. Probably the most important measurement is the amount of non-storable spent fuel. Small amounts can possibly be transshipped to other utilities; whereas, large amounts of spent fuel can potentially result in the shutdown of certain reactors. These effects are chosen to represent better the overall

Table 8. Parameter Values of the Accident Analysis Module

PARAMETER	VALUE
CUT OFF	0.0
MAXIMUM DELAY	10.0
TERMINATION TIME	99.0
B-COEFFICIENT FOR NEGATIVISM	50.0
B-COEFFICIENT FOR DELAY	0.3
C-COEFFICIENT FOR DELAY	0.1
A-COEFFICIENT FOR PERCENT REACTORS AFFECTED	1.0
B-COEFFICIENT FOR PERCENT REACTORS AFFECTED	0.0

model sensitivity. That is, using the sample discharge data, the failure point occurs prior to 1984. If the handling rate of the GE-Morris facility (on-line time of 1984) is varied, this failure point would be unaffected. Hence, combinative measurements are employed in this portion of the analysis to elucidate better parameter sensitiveness.

Table 9 gives the AFR facility information culminating from the tracking of the activities within the AFR MIS module. The activities of the MIS comprise the sample AFR MIS data. The repository, foreign discharge, and outyear discharge data are given in Chapter Two. During this portion of the sensitivity analysis, the accident analysis module is not invoked.

To analyze the effect of a change in storage pool capacity, the capacity of the main storage pool at the GE-Morris facility is varied. Table 10 condenses the results from a variance in storage pool capacity. The left column lists the different capacity values. The singular point of AFR Program objective failure is recored in the next column. The different values of percent utilization and non-storable fuel within each capacity category are recorded at the times listed in the right column (time statistics recorded). The capacities of 1500 mtu, 2000

Table 9. Sample AFR Facility Data

FACILITY (NUMBER)	ON-LINE TIME	HANDLING RATE (mtu/year)	CAPACITY (mtu)
BARNWELL (ONE)	84.5	750	1750
GE-MORRIS (TWO)	84.5	250	1100
GE-MORRIS ADD-ON (TWO)	89.2	250	1700
WEST VALLEY (THREE)	84.5	750	1700
NEW AFR (FOUR)	90.8	1000	5000

Table 10. Values from a Variance in Storage Pool Capacity

CAPACITY OF MAIN POOL (mtu)-FACILITY TWO	TIME OF AFR OBJECTIVE FAILURE	PERCENT UTILIZATION OF FACILITY	AMOUNT OF NON-STORABLE FUEL (mtu)	TIME STATISTICS RECORDED
3000	80.3	23 33 49 54 59 65	404 404 555 834 1249 2105	89.5 91.5 94.5 95.5 96.5 97.5
2000	80.3	29 42 62 68 76 82	404 404 555 834 1249 2105	89.5 91.5 94.5 95.5 96.5 97.5
1500	80.3	33 48 72 80 87 95	404 404 555 834 1249 2105	89.5 91.5 94.5 95.5 96.5 97.5
900	80.3	39 58 87 96 100 100	404 404 555 834 1249 2238	89.5 91.5 94.5 95.5 96.5 97.5
600	80.3	34 56 88 99 100 100	404 404 601 880 1316 2349	89.5 91.5 94.5 95.5 96.5 97.5

mtu, and 3000 mtu yield equal amounts of non-storable fuel since the yearly handling rate is a rather low 250 mtu/year. As expected, with the capacity set at 900 mtu, instead of 1500 mtu, a more rapid increase in both the percent utilization and amount of non-storable spent fuel occurs. Likewise, a capacity of 600 mtu causes an even quicker increase in the percent utilization and a larger accumulation of non-storable fuel.

Maintaining a constant capacity but varying the yearly handling rate of the GE-Morris facility produces the information contained in Table 11. This table presents the data in the same format as Table 10. In all cases, the reduction in percent utilization from time 88.5 to time 91.5 is due to the operation of the add-on pool at the GE-Morris facility. The top two rates (1000 mtu/year and 750 mtu/year) produce identical effects. However, further decreasing the handling rate does not necessarily produce an increase in the amount of non-storable fuel. Specifically, a handling rate of 500 mtu/year results in less non-storable fuel than a handling rate of 750 mtu/year. An in-depth review of the simulation output reveals that the higher handling rate permits the facility to achieve its maximum capacity at an earlier time; thus, terminating facility availability at an earlier time.

Table 11. Values from a Variance in the Facility Handling Rate

HANDLING RATE (mtu/Year) FACILITY TWO	TIME OF AFR OBJECTIVE FAILURE	PERCENT UTILIZATION OF FACILITY	AMOUNT OF NON-STORABLE FUEL (mtu)	TIME STATISTICS RECORDED
1000	80.3	99 100 71 100 100 100	404 404 404 971 1410 2291	86.5 88.5 91.5 95.5 96.5 97.5
750	80.3	99 100 71 100 100 100	404 404 404 971 1410 2291	86.5 88.5 91.5 95.5 96.5 97.5
500	80.3	88 100 66 100 100 100	404 404 404 673 1037 1975	86.5 88.5 91.5 95.5 96.5 97.5
250	80.3	65 87 55 91 100 100	404 404 404 834 1249 2105	86.5 88.5 91.5 95.5 96.5 97.5
100	80.3	49 66 37 51 54 58	404 404 404 1323 1860 2988	86.5 88.5 91.5 95.5 96.5 97.5

Prior to facility on-line time, discharge fuel is being stored against FCR capacity and in the temporary file. As soon as this facility becomes operational, the higher handling rate allows for a greater amount of fuel to be transferred from the FCR and the temporary tracking file into the facility. Since the non-storable amount of fuel is the gross accumulation and not a net figure, this earlier demise of Facility Two causes a slight increase in non-storable fuel.

Continual reductions in the handling rate increase the amount of non-storable fuel. A yearly handling rate of 100 mtu/year results in the largest value of fuel (2988 mtu) held in the temporary file of any of the sensitivity analyses. Withal, the facility never utilizes more than fifty-eight percent capacity.

The concluding parameter to be varied is the on-line time for Facility One. Table 12 lists the data from the different simulations in a corresponding manner. The most noticeable effects occur when the facility becomes operational prior to time 84.5. As expected, time 82.5 yields the lowest values of non-storable fuel. However, as soon as the facility reaches one hundred percent utilization, the amount of non-storable fuel increases sharply.

Table 12. Values from a Variance of the Facility On-line Time

ON-LINE TIME-FACILITY ONE	TIME OF AFR OBJECTIVE FAILURE	PERCENT UTILIZATION OF FACILITY	AMOUNT OF NON-STORABLE FUEL (mtu)	TIME STATISTICS RECORDED
86.5	80.3	N/A N/A 23 100 100 100	404 404 404 404 834 2105	83.5 85.5 87.5 91.5 95.5 97.5
85.5	80.3	N/A 11 32 100 100 100	404 404 404 404 834 2105	83.5 85.5 87.5 91.5 95.5 97.5
84.5	80.3	N/A 43 64 100 100 100	404 404 404 404 834 2105	83.5 85.5 87.5 91.5 95.5 97.5
83.5	80.3	34 50 71 100 100 100	264 264 264 264 694 1965	83.5 85.5 87.5 91.5 95.5 97.5
82.5	80.3	34 57 78 100 100 100	124 124 124 124 554 1879	83.5 85.5 87.5 91.5 95.5 97.5

EXEMPLARY RESULTS

The simulation model provides for two categories of results; intermediate output and final summary output. Both give the user an added dimension into the model. The two outputs should be employed in any analysis of a particular scenario. The sample input data employed in the parameter sensitivity analysis are also utilized in the production of the exemplary results.

The output represents the facilities, repositories, and reactors by numerical code rather than common name (e.g., reactor 25 versus Dresden-1). Numerical values are easier to handle by the computer, resulting in quicker execution times. Also, the repositories and new AFR facilities have not been assigned any literal name. Appendix E contains a conversion between numerical code and literal name for the reactors and AFR facilities. Time is represented by year and fraction rather than year and month. This is required by the GASP IV modules. The output has April, 1979 shown as 79.3; and June, 2002 represented as 102.5.

Intermediate Output

The intermediate output is accumulated throughout the simulation, providing a comprehensive portrait of the selected scenario. Figure 24 contains portions of the intermediate output from a particular simulation. The output is reviewed by reference to the different sections, as indicated by the numbers on the left-side of the figure. Each section illustrates a particular feature or features of the model. The total intermediate output can be several thousand lines in length depending upon the options chosen by the user.

Section 1: At time 79.3 Reactors 25 and 51 store fuel against their respective FCR capacity. These results indicate that neither repository nor AFR facility storage space is available. A cumulative total against FCR and the FCR capacity is given for each reactor.

Section 2: The failure point of the AFR Program objective, for this scenario, occurs at time 80.3. At this time Reactor 51 cannot store fuel either in a repository, an AFR facility, or its FCR section (FCR already contains 77 mtu from a discharge at time 79.3). This is the first occurrence of such an event. The AFR Program managers can vary the allocation of resources to

```
1   REACTOR   25. STORED   4.0 MTU AGAINST FOR AT TIME  79.3: TOTAL AGAINST FOR   4.0 MTU; FOR IS   48.7 MTU
    REACTOR   51. STORED  77.0 MTU AGAINST FOR AT TIME  79.3: TOTAL AGAINST FOR  77.0 MTU; FOR IS   82.8 MTU
     .
     .
     .
2   REACTOR   14. STORED  14.0 MTU AGAINST FOR AT TIME  80.3: TOTAL AGAINST FOR  14.0 MTU; FOR IS   70.5 MTU
    REACTOR   25. STORED   8.0 MTU AGAINST FOR AT TIME  80.3: TOTAL AGAINST FOR  12.0 MTU; FOR IS   48.7 MTU
    REACTOR NUMBER  51. CANNOT STORE  24. MTU AT  80.3
     .
     .
     .
3   REACTOR NUMBER 300. CANNOT STORE  18. MTU AT  81.3
    REACTOR NUMBER 301. CANNOT STORE  16. MTU AT  81.3
    REACTOR NUMBER 302. CANNOT STORE  16. MTU AT  81.3
    REACTOR   14. STORED  20.0 MTU AGAINST FOR AT TIME  81.3: TOTAL AGAINST FOR  34.0 MTU; FOR IS   70.5 MTU
    REACTOR   16. STORED  26.0 MTU AGAINST FOR AT TIME  81.3: TOTAL AGAINST FOR  26.0 MTU; FOR IS  109.2 MTU
     .
     .
     .
4   FACILITY NUMBER  2 HAS INCREASED CAPACITY BY 1100. MTU AT TIME =  84.5
        MAX CAPACITY IS    1100.0; HANDLING RATE IS    250.0 MTU PER YEAR
    REACTOR   16. STORED 107.0 MTU AT TIME    84.5 IN FACILITY    2 TO REGAIN FOR
    REACTOR   51. STORED  77.0 MTU AT TIME    84.5 IN FACILITY    2 TO REGAIN FOR
    REACTOR   25. STORED  44.0 MTU AT TIME    84.5 IN FACILITY    2 TO REGAIN FOR
    REACTOR   14. REMOVED    22.0 MTU FROM ITS AR POOL AT TIME    85.
            THIS AMOUNT WAS DICTATED BY THE HANDLING RATE OF FACILITY   2
            THIS REACTOR STILL IS UTILIZING    45.4 PERCENT OF ITS FOR CAPACITY
    FACILITY NUMBER 1 HAS INCREASED CAPACITY BY 1750. MTU AT TIME =  84.5
        MAX CAPACITY IS    1750.0; HANDLING RATE IS    750.0 MTU PER YEAR
    REACTOR  159. STORED  36.0 MTU AT TIME    84.5 IN FACILITY    1 TO REGAIN FOR
     .
     .
     .
    REACTOR  189. STORED   5.0 MTU AT TIME    84.5 IN FACILITY    1 TO REGAIN FOR
    REACTOR   51. HAS REMOVED    24.0 MTU FROM TEMPORARY STORAGE
            FUEL STORED IN FACILITY NUMBER    1 AT TIME    84.5
    REACTOR   14. HAS REMOVED    40.0 MTU FROM TEMPORARY STORAGE
            FUEL STORED IN FACILITY NUMBER    1 AT TIME    84.5
```

Figure 24. Intermediate Output from Simulation Model

```
    .
    .
    .
┌── FACILITY NUMBER  2 HAS EXCEEDED ITS HANDLING RATE AT TIME  85.3
│
│   *****REACTOR  300. STORED  31.0 MTU IN FACILITY  1 AT TIME  85.3
│   *****REACTOR  301. STORED  33.0 MTU IN FACILITY  3 AT TIME  85.3
│   *****REACTOR  302. STORED  31.0 MTU IN FACILITY  1 AT TIME  85.3
│
│   FACILITY NUMBER  2 HAS EXCEEDED ITS HANDLING RATE AT TIME  85.3
5 ─┤
│   *****REACTOR    8. STORED  18.0 MTU IN FACILITY  1 AT TIME  85.3
│   *****REACTOR   14. STORED  20.0 MTU IN FACILITY  1 AT TIME  85.3
│   *****REACTOR   15. STORED  27.0 MTU IN FACILITY  1 AT TIME  85.3
│
│   FACILITY NUMBER  1 HAS EXCEEDED ITS HANDLING RATE AT TIME  85.3
│
└── *****REACTOR   16. STORED  27.0 MTU IN FACILITY  3 AT TIME  85.3
    .
    .
    .
┌── *****REACTOR  300. STORED  48.0 MTU IN FACILITY  2 AT TIME  86.3
│   *****REACTOR  301. STORED  46.0 MTU IN FACILITY  3 AT TIME  86.3
│   *****REACTOR  302. STORED  46.0 MTU IN FACILITY  1 AT TIME  86.3
│   *****REACTOR    8. STORED  21.0 MTU IN FACILITY  2 AT TIME  86.3
│   *****REACTOR   14. STORED  20.0 MTU IN FACILITY  1 AT TIME  86.3
│   *****REACTOR   15. STORED  27.0 MTU IN FACILITY  1 AT TIME  86.3
6 ─┤ *****REACTOR   16. STORED  27.0 MTU IN FACILITY  1 AT TIME  86.3
│   *****REACTOR   25. STORED   8.0 MTU IN FACILITY  2 AT TIME  86.3
│   *****REACTOR   42. STORED  27.0 MTU IN FACILITY  2 AT TIME  86.3
│   *****REACTOR   51. STORED  17.0 MTU IN FACILITY  1 AT TIME  86.3
│   *****REACTOR   67. STORED  19.0 MTU IN FACILITY  1 AT TIME  86.3
│   *****REACTOR   85. STORED  21.0 MTU IN FACILITY  3 AT TIME  86.3
└── *****REACTOR  106. STORED  11.0 MTU IN FACILITY  3 AT TIME  86.3
    .
    .
    .
┌── FOLLOWING STATISTICS AT TIME   87.5
│
│                              **STATISTICS FOR TIME-PERSISTENT VARIABLES**
│            MEAN        STD DEV      MINIMUM        MAXIMUM      TIME INTERVAL   CUR. VALUE
│   FACIL1   0.4513E+00  0.8022E-01   0.0            0.6383E+00   0.3000E+01      0.6383+00
│
│                              **STATISTICS FOR TIME-PERSISTENT VARIABLES**
7 ─┤         MEAN        STD DEV      MINIMUM        MAXIMUM      TIME INTERVAL   CUR. VALUE
│   FACIL2   0.4573E+00  0.1718E+00   0.0            0.7455E+00   0.3000E+01      0.7455E+00
│
│                              **STATISTICS FOR TIME-PERSISTENT VARIABLES**
│            MEAN        STD DEV      MINIMUM        MAXIMUM      TIME INTERVAL   CUR. VALUE
└── FACIL3   0.2264E+00  0.9873E-01   0.0            0.4432E+00   0.3000E+01      0.4432E+00
```

Figure 24. (continued)

```
.
.
.
REPOSITORY NUMBER 1 HAS INCREASED CAPACITY BY 41000.0 MTU AT TIME=  97.6
    MAX CAPACITY IS  41000.0; HANDLING RATE IS  1800.0 MTU PER YEAR

REACTOR   43 STORED  80.0 MTU AT TIME  97.6 IN REPOSITORY  1 TO REGAIN FOR

REACTOR  128 STORED  84.0 MTU AT TIME  97.6 IN REPOSITORY  1 TO REGAIN FOR
.
.
.
REACTOR  173 STORED 102.0 MTU AT TIME  97.6 IN REPOSITORY  1 TO REGAIN FOR

8 REACTOR  175 STORED 102.0 MTU AT TIME  97.6 IN REPOSITORY  1 TO REGAIN FOR

REACTOR  137 REMOVED  39.0 MTU FROM ITS AR POOL AT TIME  97.6
        THIS AMOUNT WAS DICTATED BY THE HANDLING RATE OF REPOSITORY  1
        THIS REACTOR STILL IS UTILIZING  25.0 PERCENT OF ITS FOR CAPACITY

REPOSITORY NUMBER 1 HAS EXCEEDED ITS HANDLING RATE AT TIME  98.3

REACTOR NUMBER   5. CANNOT STORE  9. MTU at  98.3 IN ANY REPOSITORY

REACTOR    5. STORED  9.0 MTU AGAINST FOR AT TIME  98.3: TOTAL AGAINST FOR  9.0 MTU; FOR IS  102.9 MTU
.
.
.
*****REACTOR  290. STORED  389.0 MTU IN REPOSITORY  1 AT TIME  100.3
*****REACTOR  291. STORED  389.0 MTU IN REPOSITORY  1 AT TIME  100.3
*****REACTOR  292. STORED  389.0 MTU IN REPOSITORY  1 AT TIME  100.3
*****REACTOR  293. STORED  389.0 MTU IN REPOSITORY  1 AT TIME  100.3

REPOSITORY NUMBER  1 HAS EXCEEDED ITS HANDLING RATE AT TIME   100.3

9 REACTOR NUMBER 294. CANNOT STORE 389. MTU AT 100.3 IN ANY REPOSITORY

*****REACTOR  294. STORED  389.0 MTU IN FACILITY  4 AT TIME  100.3

REPOSITORY NUMBER 1 HAS EXCEEDED ITS HANDLING RATE AT TIME   100.3

REACTOR NUMBER 295. CANNOT STORE 389. MTU AT 100.3 IN ANY REPOSITORY

*****REACTOR  295. STORED  389.0 MTU IN FACILITY  4 AT TIME  100.3
```

Figure 24. (continued)

```
.
.
.
FACILITY    2 REMOVED  108.0 MTU AND PLACED FUEL INTO REPOSITORY  1 AT TIME  101.7
        AMOUNT WAS LIMITED BY REPOSITORY HANDLING RATE OR AVAILABLE CAPACITY

FACILITY    3 REMOVED  750.0 MTU AND PLACED FUEL INTO REPOSITORY  2 AT TIME  101.7
        AMOUNT WAS LIMITED BY FACILITY HANDLING RATE

FACILITY    2 REMOVED  142.0 MTU AND PLACED FUEL INTO REPOSITORY  2 AT TIME  101.7
        AMOUNT WAS LIMITED BY FACILITY HANDLING RATE
10
FACILITY    4 REMOVED  908.0 MTU AND PLACED FUEL INTO REPOSITORY  2 AT TIME  101.7
        AMOUNT WAS LIMITED BY REPOSITORY HANDLING RATE OR AVAILABLE CAPACITY

*****REACTOR  290. STORED  441.0 MTU IN REPOSITORY  1 AT TIME  102.3
*****REACTOR  291. STORED  441.0 MTU IN REPOSITORY  1 AT TIME  102.3
*****REACTOR  292. STORED  441.0 MTU IN REPOSITORY  1 AT TIME  102.3
*****REACTOR  293. STORED  441.0 MTU IN REPOSITORY  1 AT TIME  102.3
.
.
.
REPOSITORY NUMBER  3 HAS INCREASED CAPACITY BY    0.0 MTU AT TIME=  108.6
    MAX CAPACITY IS  69000.0; HANDLING RATE IS  6000.0 MTU PER YEAR

REACTOR  298 HAS REMOVED   93.0 MTU FROM TEMPORARY STORAGE
        FUEL STORED IN REPOSITORY NUMBER  3 AT TIME   0.0

REACTOR  299 HAS REMOVED 1253.0 MTU FROM TEMPORARY STORAGE
        FUEL STORED IN REPOSITORY NUMBER  3 AT TIME   0.0
11
*****REACTOR  290. STORED  502.0 MTU IN REPOSITORY  4 AT TIME  109.3
*****REACTOR  291. STORED  502.0 MTU IN REPOSITORY  4 AT TIME  109.3
*****REACTOR  292. STORED  502.0 MTU IN REPOSITORY  4 AT TIME  109.3

REPOSITORY NUMBER  4 HAS EXCEEDED ITS HANDLING RATE AT TIME   109.3

*****REACTOR  293. STORED  502.0 MTU IN REPOSITORY  3 AT TIME  109.3
```

Figure 24. (continued)

effect a different scenario in anticipation of eliminating this failure. At this time, Reactor 25 stores an additional 8 mtu against FCR, bring the total amount of fuel against FCR to 12 mtu.

Section 3: Reactors 300, 301, and 302 represent the foreign discharges. There is no FCR capacity associated with these pseudo-reactors. If neither repository nor AFR space is available, then the discharges cannot be stored. However, the fuel is tracked in the temporary file.

Section 4: The first AFR facility to come on-line is Facility 2 (GE-Morris). This event occurs at time 84.5. The facility capacity is 1100 mtu and the handling rate is 250 mtu/year. Afterwards, spent fuel is removed from the FCR sections of any reactor storing fuel against FCR. The priority is that the FCR with the greatest percentage of utilized capacity transfers fuel first. Notice that Reactor 14 can remove only 22 mtu from its FCR due to the handling rate of Facility Two. For the next year, Facility Two cannot receive fuel from any source. Facility One comes on-line at 84.5 and receives fuel from FCR space. After all reactors have regained their FCR capabilities, an examination is performed on the temporary file. Any fuel tracked in this file is removed and stored in Facility One. This process of storing fuel in the temporary file is optional and can simulate any desired storage method or methods (e.g., transshipment).

Section 5: Shipment priority to the AFR facility is ship to the nearest facility if space is available. If space is not available, ship the discharge to the next nearest and so on. For Reactor 300 the nearest facility is Facility 2; however, Facility Two has exceeded its yearly handling rate. The next nearest facility is Facility 1 and the fuel is stored. In an analogous fashion, fuel is stored in the indicated facilities. At time 85.3, Facilities 1 and 2 have exceeded their handling rates.

Section 6: All three AFR facilities have available storage space at time 86.3.

Section 7: Each year statistics on the percent utilization of the operating storage facilities are outputted. At time 87.5 Facility One is utilizing 64 percent of its capacity; Facility Two, 75 percent; and Facility Three, 44 percent. The time interval for these statistics is three years, as said interval should be. Normally, GASP IV would have employed a time interval starting at time 78.0; but, a program modification adjusts this calculation. The ease of performing this modification emphasizes the correctness of selecting GASP

IV for this simulation model.

Section 8: Repository One comes on-line at time 97.6. In a manner kindred to the operation of an AFR facility, the simulation proceeds. At time 98.3, spent fuel from Reactor 5 attempts to be stored in a repository; however, none are available. The next choice is an AFR facility; again, none are available. The spent fuel is finally stored against FCR capacity. Here, a specific message indicating the unavailability of AFR space is not printed. Whenever fuel is stored in the FCR or temporary file, repository and AFR space have been examined and discovered lacking. Since a specific line indicates no repository space and another line of output indicates that FCR storage is employed, AFR examination is implied. This reduces the output to a manageable length.

Section 9: This section depicts the shipment order. Discharges are stored in Repository One until the handling rate is exceeded. Then, since no other repositories are operational, the discharges are stored in Facility 4.

Section 10: At time 101.7 (corresponding to August, 2001) spent fuel from AFR facilities is shipped and stored in the repositories. This exchange continues until either the handling rate of the repository or facility, or the capacity of the repository is exceeded. By the time

102.3, the yearly repository handling rate has been reset and discharges from individual reactors are shipped directly to Repository Two.

Section 11: Repository Three increases its handling rate from 1800 mtu/year to 6000 mtu/year. Next, fuel is removed from the temporary file. At this point there is no fuel being held against FCR nor any fuel being tracked in the temporary file. Reactor discharges continue to be stored in the repositories until the end-of-simulation.

The preceeding output is produced during the simulation, as each action occurs. Summary output is accumulated at the end of the simulation run. The summary output is reviewed in the next section.

Final Summary Output

This output summarizes the entire simulation. The mean, standard deviation, minimum and maximum value, and current value of the utilization for each AFR facility and repository is given in Figure 25. Notice that each AFR facility has a current utilization value of zero, indicating all spent fuel has been removed from the facilities and transferred into the repositories. Figure

End-of-Simulation Statistics-

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
FACIL1	0.4341E+00	0.4367E+00	0.0	0.1000E+01	0.3950E+02	0.0

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
FACIL2	0.5791E+00	0.3012E+00	0.0	0.9996E+00	0.3950E+02	0.0

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
FACIL3	0.3844E+00	0.4368E+00	0.0	0.9997E+00	0.3950E+02	0.0

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
FACIL4	0.3486E+00	0.3799E+00	0.0	0.1000E+01	0.3320E+02	0.0

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
REPOS1	0.7486E+00	0.3601E+00	0.0	0.1000E+01	0.2640E+02	0.1781E+00

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
REPOS2	0.1173E+00	0.5617E+01	0.0	0.1781E+00	0.2340E+02	0.1781E+00

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
REPOS3	0.1855E+00	0.8386E+01	0.0	0.2408E+00	0.2040E+02	0.2408E+00

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
REPOS4	0.4029E+00	0.3063E+00	0.0	0.8228E+00	0.1740E+02	0.8228E+00

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
REPOS5	0.4533E-01	0.1464E-01	0.0	0.6193E-01	0.1440E+02	0.6193E-01

FACILITY NUMBER 1 HAD 78 SHIPMENTS

FACILITY NUMBER 2 HAD 94 SHIPMENTS

FACILITY NUMBER 3 HAD 64 SHIPMENTS

FACILITY NUMBER 4 HAD 188 SHIPMENTS

REPOSITORY NUMBER 1 HAD 148 SHIPMENTS

REPOSITORY NUMBER 2 HAD 38 SHIPMENTS

REPOSITORY NUMBER 3 HAD 79 SHIPMENTS

REPOSITORY NUMBER 4 HAD 102 SHIPMENTS

REPOSITORY NUMBER 5 HAD 8 SHIPMENTS

Figure 25. Final Summary Output from Simulation Model

25 also includes the number of spent fuel shipments for each facility and repository.

Perhaps the best method to summarize a particular scenario is through the use of graphs. The output provides for a series of graphs of supply versus demand for 1) each AFR facility, 2) a combination of all AFR facilities, 3) each repository, and 4) a combination of all repositories. Figure 26 is the graph for the new AFR facility number one; Figure 27, the summation of all AFR facilities; and Figure 28, the summation of all repositories. Besides the supply and demand curves, a heavy vertical line indicates the AFR Program objective failure point. For this simulation run, the failure point is April, 1980. The shape of the AFR curves shows the initial receipt of spent fuel from the reactors followed by the transfer of this fuel to the repositories. The dip in the demand curves for the AFR facilities is caused by the shipment of fuel from the facility to the repository. Initially, this amount of fuel adds to the handling rate of both AFR site and repository. Next, spent fuel is shipped directly to the repository, further consuming the repository handling rate. When the repository rate is exceeded, discharged fuel is then directed to the AFR facilities, increasing their demand. Finally, additional

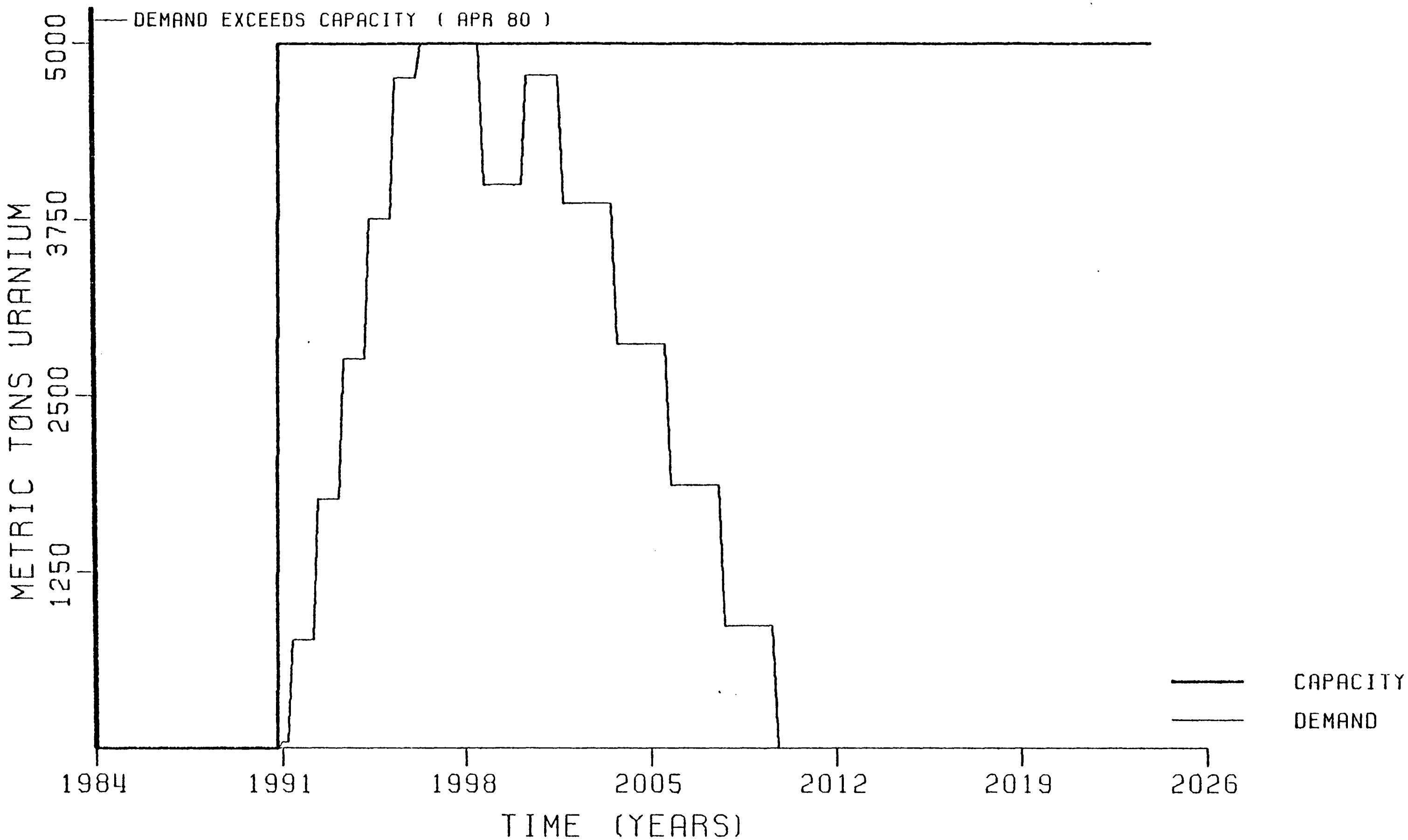


Figure 26. Storage Utilization for New AFR Facility Number 1
Shipment Mode to AFR: N
Based on Sample Input Data

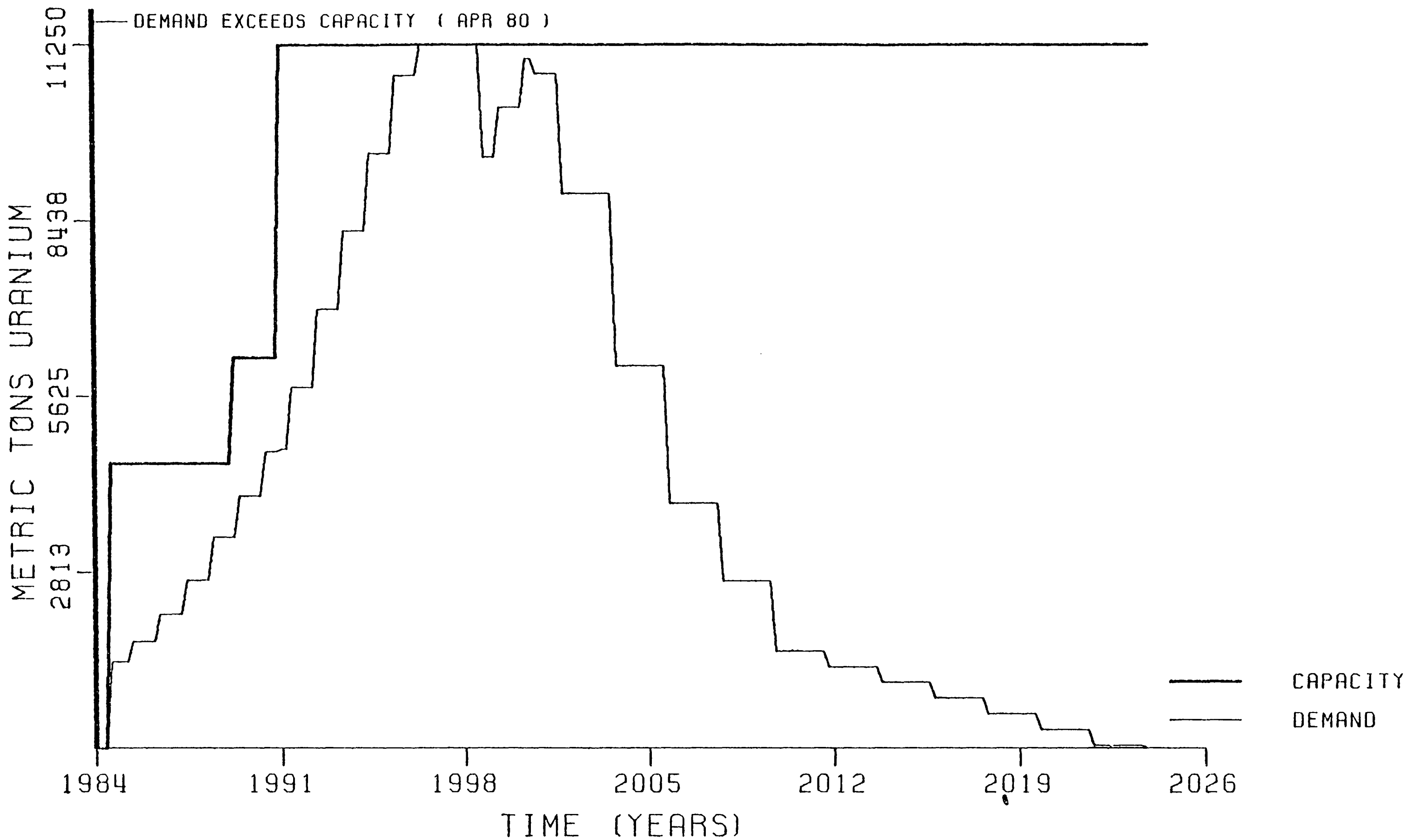


Figure 27. Storage Utilization for all AFR Facilities
Shipment Mode to AFR: N
Based on Sample Input Data

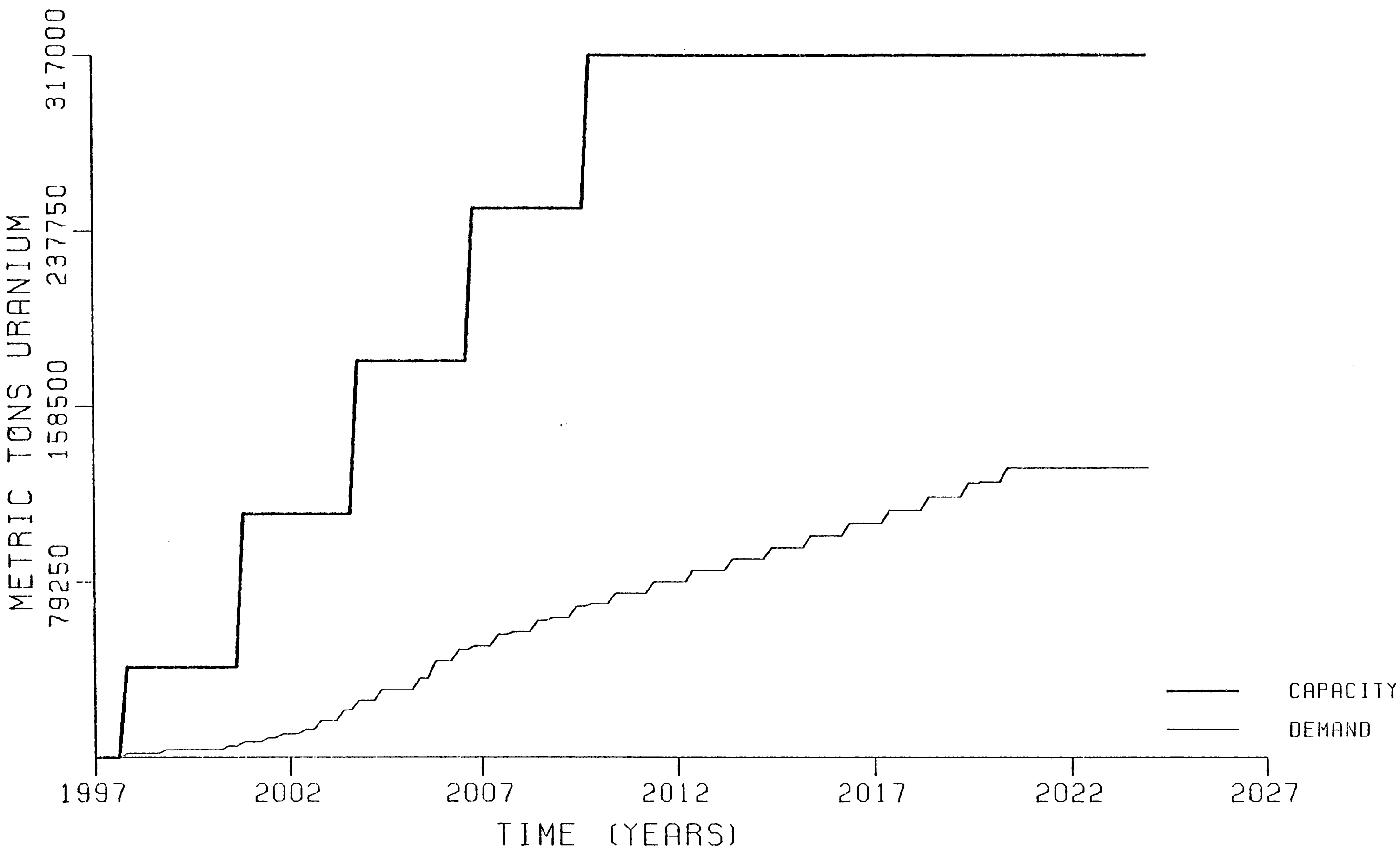


Figure 28. Storage Utilization for all Repositories
Shipment Mode to Repository: N
Based on Sample Input Data

repositories become operational and the AFR demand decreases to zero.

Now that the exemplary output has been discussed, the results of a simulation based upon the most accurate data available as of August, 1980 are summarized in the following section.

RESULTS DERIVED FROM AUGUST 1980 INPUT DATA

A review of the simulation results based upon the best-case input data accumulated in August of 1980 is given in this section. Appendix F contains a detailed accounting of this input data, excluding the AFR MIS activity listing. An activity listing is over two hundred pages and, therefore, not included in the appendix. Instead, the output from the interrelationship tracking program and the milestone extraction program is given. This output contains the AFR facilities on-line times, yearly handling rates, capacities, and amounts of spent fuel being stored on-site. The accident analysis module is not used during this simulation. Both the FCR and the temporary file storage options are permitted. The selective mode of shipment, based upon NERC region, is chosen for discharge shipment from the reactor to the AFR site. If the specified facility is unavailable, the discharge is transferred to the nearest AFR facility. Discharges are shipped to the nearest repository.

Employing the above selected options, the simulation model is exercised. The simulation requires twenty-three seconds of computer CPU time and less than 384 kilobytes of CPU memory. A total cost of approximately ten dollars is incurred. Sixty-five percent of this value covers the cost of over nine thousand lines of printed output.

The results indicate that the AFR Program cannot meet its objective at time 80.3 (i.e., April 1980) due to the discharge from Reactor 51 (Oconee-3). An anaylsis of this failure point, including an explanation as to why Oconee-3 is still operating, is given in Chapter Six. The first storage site becomes operational at time 84.5. Prior to this time, 404 mtu of spent fuel cannot be stored and is tracked within the temporary file. Ninety-four percent of the 404 mtu of non-storable fuel is attributed to foreign spent-fuel requirements. A total of ten different reactors are required to store fuel against their FCR capacity before time 84.5.

Between time 84.5 and time 97.5 all discharges are stored either in an AFR facility or in the FCR sections of on-site pools; hence, no additional non-storable fuel is accumulated during this interval. A total of 2711 mtu of fuel is stored against FCR between times 84.5 and 97.5. All AFR facilities, except new AFR number 2, reach maximum capacity at time 96.5. The first repository begins accepting fuel at time 97.6. Due to a selected option, fuel stored against FCR space is shipped to this repository until its handling rate is exceeded. At time 98.7, the repository handling rate is reset and fuel from the AFR facilities is relocated into the repository. This relocation continues until the yearly handling rate of the repository is maximized. AFR storage space is once again available. This availability permits the storage of spent fuel in the AFR sites until the associated handling rates are exceeded.

Due to the large amount of projected spent fuel discharges after time 98.5, the handling rates of Repository One and the AFR facilities are quickly maximized. After the FCR sections of the at-reactor storage pools become full, non-storable fuel accumulates in the temporary file. At time 100.5 a total of 2779 mtu of non-storable fuel is obtained; at time 101.5, 5317 mtu; and at time 102.5, 7963 mtu. By time 102.5, Repository Two starts operating and the handling rate for Repository

One increases from 1800 mtu/year to 6000 mtu/year. This additional storage capacity, in conjunction with the future capacity provided by Repositories Three, Four, and Five, avoids any additional accumulation of non-storable fuel throughout the simulation.

Figures 29 and 30 are the graphical representations of total AFR facility and total repository supply and demand. Figure 31 shows the final statistics on the percent utilization of each storage site. The complete series of output graphs is supplied in Appendix F.

An interpretation of these results is given in Chapter Six. This next chapter also contains overall model performance conclusions.

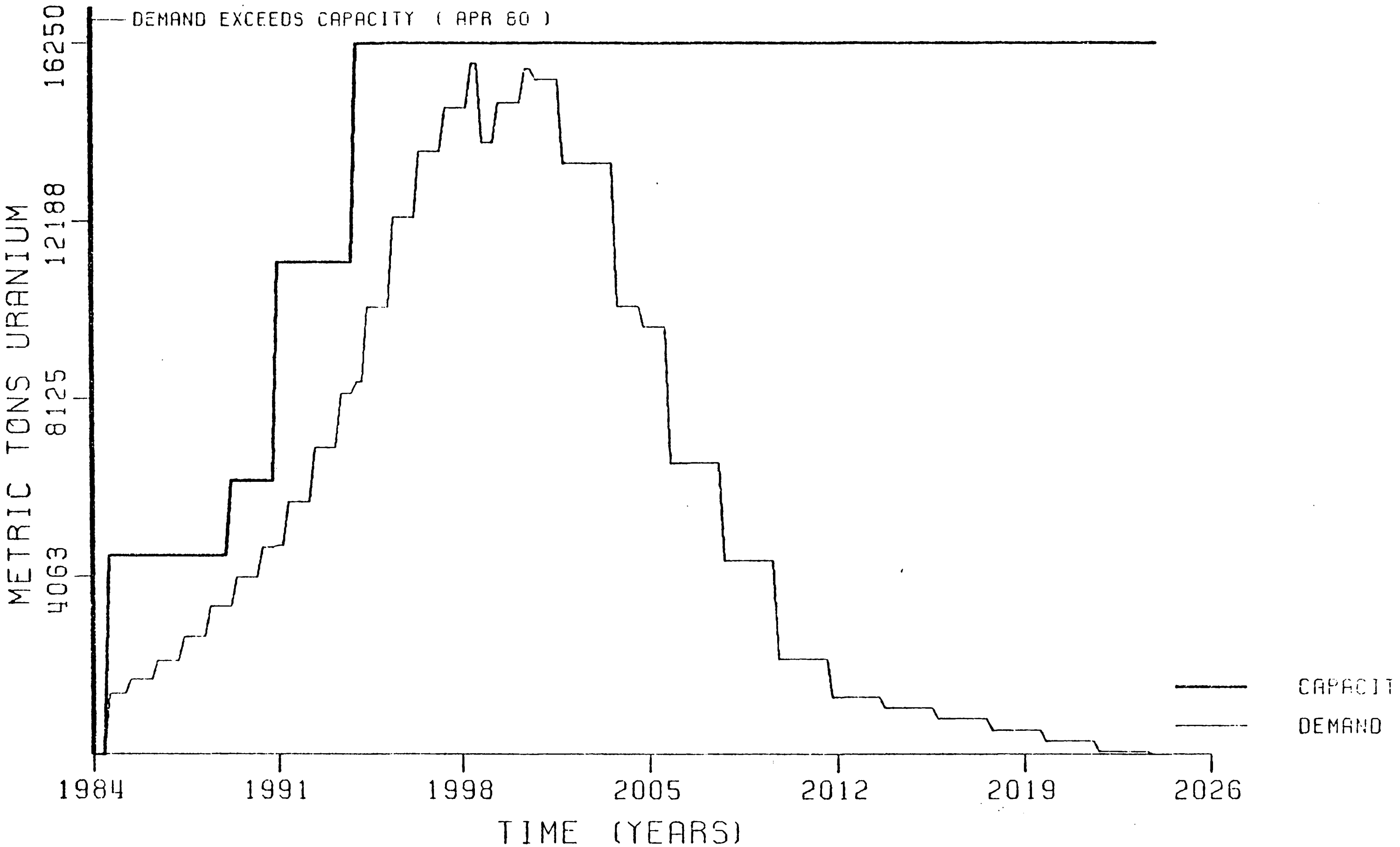


Figure 29. Storage Utilization for all AFR Facilities
Shipment Mode to AFR: S
Based on Best-case Input Data

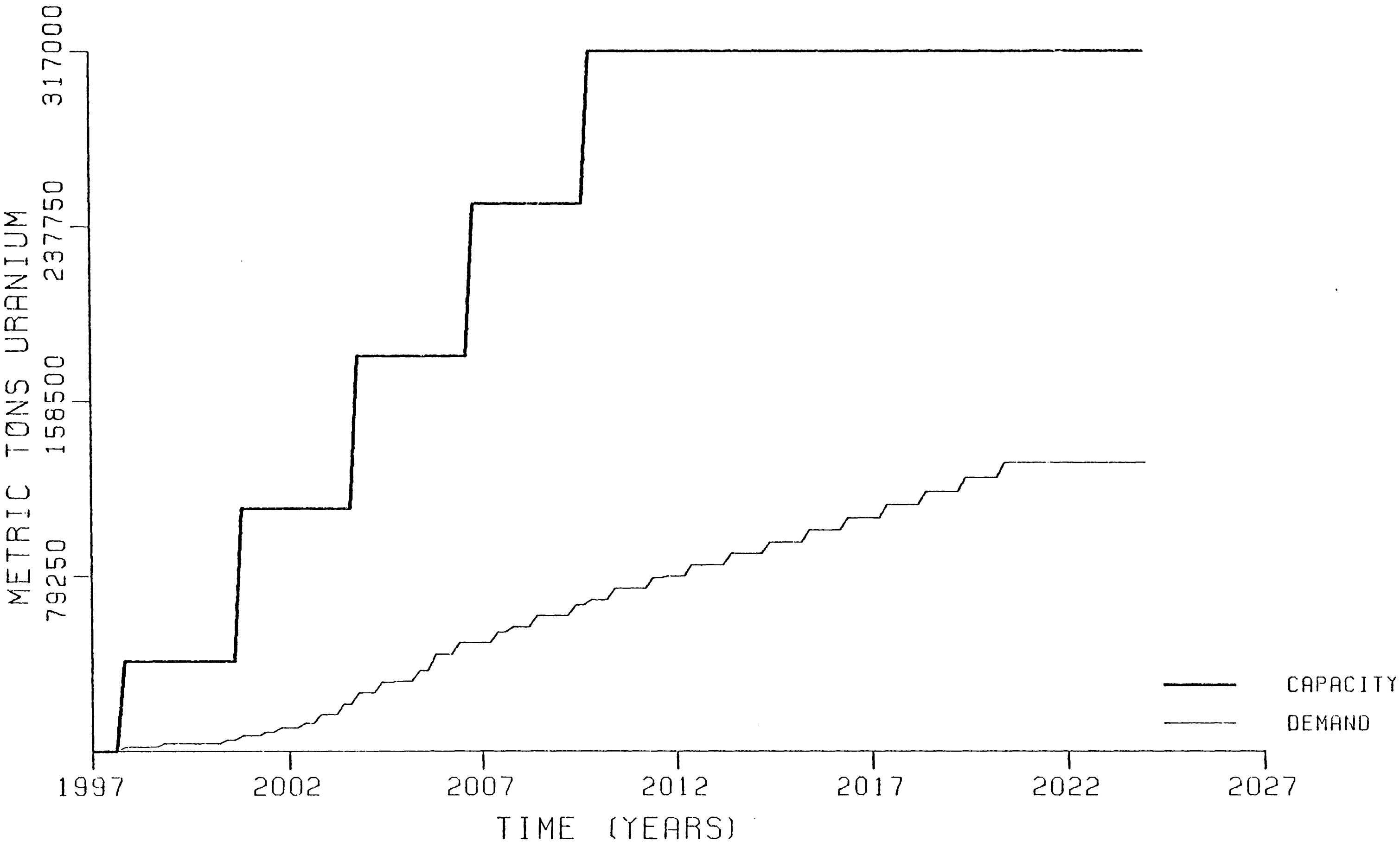


Figure 30. Storage Utilization for all Repositories
Shipment Mode to Repository: N
Based on Best-case Input Data

STATISTICS FOR TIME-PERSISTENT VARIABLES

	MEAN	STD DEV	MINIMUM	MAXIMUM	TIME INTERVAL	CUR. VALUE
FACIL1	0.4270E+00	0.4312E+00	0.0	0.9994E+00	0.3950E+02	0.0
FACIL2	0.5886E+00	0.2912E+00	0.0	0.9996E+00	0.3950E+02	0.0
FACIL3	0.3840E+00	0.4311E+00	0.0	0.9985E+00	0.3950E+02	0.0
FACIL4	0.3911E+00	0.3898E+00	0.0	0.9996E+00	0.3320E+02	0.0
FACIL5	0.2978E+00	0.3552E+00	0.0	0.9996E+00	0.3020E+02	0.0
REPOS1	0.7547E+00	0.3543E+00	0.0	0.1000E+01	0.2640E+02	0.1000E+01
REPOS2	0.1541E+00	0.5425E-01	0.0	0.2020E+00	0.2340E+02	0.2020E+00
REPOS3	0.1708E+00	0.7608E-01	0.0	0.2213E+00	0.2040E+02	0.2213E+00
REPOS4	0.4002E+00	0.3042E+00	0.0	0.8184E+00	0.1740E+02	0.8184E+00
REPOS5	0.4533E-01	0.1464E-01	0.0	0.6193E-01	0.1440E+02	0.6193-01

FACILITY NUMBER 1 HAD 62 SHIPMENTS
FACILITY NUMBER 2 HAD 119 SHIPMENTS
FACILITY NUMBER 3 HAD 64 SHIPMENTS
FACILITY NUMBER 4 HAD 171 SHIPMENTS
FACILITY NUMBER 5 HAD 151 SHIPMENTS

REPOSITORY NUMBER 16 HAD 158 SHIPMENTS
REPOSITORY NUMBER 17 HAD 28 SHIPMENTS
REPOSITORY NUMBER 18 HAD 30 SHIPMENTS
REPOSITORY NUMBER 19 HAD 103 SHIPMENTS
REPOSITORY NUMBER 20 HAD 8 SHIPMENTS

Figure 31. Final Statistics based on Best-case Input Data

CHAPTER SIX - CONCLUSIONS

Conclusions based upon model operation and the exemplary results are given in the first section of this chapter. The remaining section ratiocinates upon the results based on the best available data as of August, 1980.

GENERIC CONCLUSIONS

The simulation model is verified through the mechanisms described in Chapter Three and detailed in Appendix B. Except for the accident analysis module, this verification process confirms the accuracy of the model. There are no probabilities associated with this descriptive simulation. A discharge of spent fuel is either stored in a repository, an AFR facility, or the FCR portion of the at-reactor pool or the fuel is not stored. Whenever hand calculations confirm this series of events, the correctness of the model is ensured.

The sensitivity analysis performed upon the accident analysis module shows a definite relationship between the yearly accident rate and the failure point of the AFR Program objective. A ninety-five percent confidence interval varies from minus 1.7 years to plus 1.68 years. This interval results from statistical variations and not from a lack of correlation between the two parameters.

An analysis of variations in the yearly handling rate and the capacity of Facility Two indicates a practical relation between these two parameters. A large facility capacity and a low handling rate results in a low overall utilization, yielding a waste of storage resource. A large handling rate and a low capacity causes rapid fuel storage into the facility. The facility quickly maximizes capacity and goes off-line. As the amount of non-storable fuel indicates, a low facility capacity is definitely not desirable.

Bringing Facility One on-line at successively earlier times reduces the overall amount of non-storable fuel. Due to governmental restrictions this action is highly unlikely. In any case, the failure point remains fixed at time 80.3. From the perspective of a program manager, this insensitivity of the failure point indicates that the only method with which to achieve the AFR Program objective is some form of transshipment.

If extremely optimistic estimates of the on-line time and capacity of the AFR facilities are utilized, then the total elimination of the failure point is possible. In such a case, the accident module can be employed to judge the effects that different yearly accident rates have on the percent utilization.

In summary, the simulation model is found sensitive, to some degree, to each of the varied parameters. The only measurement not effected by the variances is the failure point. Some parameters have a cutoff value, above or below which changes have minor effects upon the total simulated scenario. No one parameter is found to have an overriding influence upon the simulation results.

A variance in repository characteristics is expected to yield comparable analyses. Changes in the discharge data are viewed through the accident analysis section since the module acts upon the individual discharges.

REALISTIC CASE CONCLUSIONS

The first reactor which cannot successfully store spent fuel is Oconee-3. The simulation model predicts that this event occurs in April 1980. At the time of this writing (Summer, 1980), Oconee-3 is still operating. The on-site storage pool is ninety-three percent utilized, including fuel stored against FCR. Oconee-3 is scheduled for refueling in November of 1980. In order to ensure storage space for the discharge from the refueling, transshipment requests to the Mcquire-1 reactor are presently being processed. If approval for transshipment is not realized by mid-November 1980, then Oconee-3 will remain inoperable until some interim storage agreement can be approved.

The discharge data obtained from the Stoller Database is listed by year. In the model a specific time is required for this simulated event. The value of April of each year is selected since electrical requirements ebb during this period. When the simulation reports a failure point of time 80.3, this does not necessarily mean that the particular reactor is shutting down on the twelfth day in April. Rather, the result flags a critical area which requires further investigation by the program managers. In the case of Oconee-3, its actual discharge occurs in the Fall of 1980. The simulation does indicate trouble for the Oconee-3 reactor in 1980, and further investigation reveals that some form of transshipment must be approved to avoid an extended refueling shutdown.

Until the first AFR facilities come on-line in June of 1984, the non-storable fuel is accumulated from foreign discharges, except for twenty-four mtu from Oconee-3. The political nature of the decision to accept foreign spent fuel makes definitive statements on this subject difficult. Obviously, if this fuel reaches the borders of the United States in 1981, some form of temporary storage is required.

Between 1984 and 1996, all discharges are stored either in an AFR facility or the FCR section of at-reactor pools. The large amount of fuel (2711 mtu) accumulated within the FCR sections of the pools is an ominous sign. When Repository One comes on-line in 1997, the repository accepts the fuel held against FCR. This large amount of fuel ties up the handling section of the repository for some time. While fuel is being transferred from the FCR sections to the repository, numerous reactors continue to discharge fuel. Most of this fuel has to be tracked within the temporary file. Approximately five years, starting in 1997, are required to relieve the backlog of fuel held against FCR and in the temporary file. Not until 2003 is the need to track fuel in the temporary file relaxed.

The analysis of this particular scenario is disheartening. The large accumulation of non-storable fuel (7963 mtu) has to be transshipped in some manner to avoid reactor shutdowns. Shipping cask capacity, as of 1980, is twenty-eight mtu of pressurized water reactor (PWR) fuel.[6] Approximately three years are required for the construction of a cask of approved design. Up to eight years is required for the construction of casks of neoteric design. The transshipment of 7963 mtu of fuel is going to be interesting, especially when yearly discharges to available AFR facilities are competing for cask usage. And if the Nuclear Regulatory Commission (NRC) ever requires each reactor to maintain FCR, the storage problems become overwhelming.

REFLECTIONS

The glum results of the aforementioned scenario indicate the desperate need for some type of model of spent fuel disposition. The developed simulation model satisfies this need. For now the DOE managers possess a tool with which to analyze any disposition scenario. Licensing activities can be hastened; the time required for environmental impact statements can be reduced; AFR facility handling rates can be increased; and so on. The impact upon spent fuel disposition of each of these variations can be inexpensively analyzed in the hope of determining an acceptable, realistic waste scenario.

A SCENARIO THAT WORKS

The value of the model is demonstrated by determining a workable solution to the task of providing adequate storage space. The model is employed on an iterative basis. A scenario is posed, simulated, and analyzed. The author decides on any improvements and/or modifications, adjusts the scenario, and performs another simulation. After several iterations, the result is a finely honed storage supply scenario which provides adequate storage capacity for all the reactor discharges given in appendix F (best-case data). The basic approach is to reduce the time required for licensing and other regulatory functions. Except where specifically noted, the capacities and handling rates remain the same as those given in the best-case data. The following paragraphs delineate the required scenario.

One: Permit, in November 1980, the transshipment of fuel from Oconee-3 to another at-reactor storage pool.

Two: The GE-Morris plant begins storage operations as an AFR facility in early 1981. The add-on storage pool is started in early 1987 with a handling rate of 500 mtu/year.

Three: In January 1982, begin storage operation of the West Valley facility.

Four: The AGNS storage facility starts accepting fuel in mid-1984.

Five: Construct three new AFR facilities. The first starts operation in mid-1988; the second, mid-1990; and the third, mid-1992.

Six: Complete construction and begin operation of the first repository in June 1994. The required handling rate is 3000 mtu/year.

Seven: The second repository requires a handling rate of 6000 mtu/year and comes on-line in early 1997.

Eight: Bring into operation, at three year intervals starting in 2003, three additional repositories.

Although reducing the time to accomplish a licensing or design activity via the simulation is straightforward; an actual, real-world reduction in time may not be feasible. However, given the proper motivation, there does not appear to be any reason why this scenario cannot be realized.

CHAPTER SEVEN - RECOMMENDATIONS

This work lays the foundation for a simulation model which encompasses the supplies and demands associated with all sources of nuclear wastes. The present model only tracks spent fuel from commercial reactors. Other sources of supply include defense wastes, high level wastes, low level wastes, and transuranic wastes. Each category comes under the supervision of a separate program office. Since the ultimate disposition of all these waste products is currently projected as the geologic repositories, the simulation model can be enhanced to track all such wastes.

The ONWI is developing a MIS containing detailed repository data. This MIS should contain all activities associated with repository design, licensing, construction, and operation. In turn, these activities should be linked via predecessor/successor relationships in order to track program progress in a manner analogous with the AFR MIS module. This will provide a more comprehensive and flexible source of repository data.

Currently, the policy of the United States prohibits the reprocessing of spent nuclear fuel. When this policy changes, spent fuel will be sent to the reprocessing plants not only from the individual reactors but from the AFR facilities. The model should be expanded to simulate the reprocessing shipments.

Transshipment of spent fuel is a critical need if individual reactor shutdowns are to be avoided. Presently, approval for transshipment of fuel is done on a case-by-case basis. As more and more shipments are required, some specific guidelines should emerge. When such guidelines are formulated, the simulation model will incorporate said guidelines and accurately model this important stopgap storage technique.

Finally, the model is expected to find extensive utilization by the AFR Program managers. As the model is exercised under real world conditions, various additions will inevitably be needed. During all phases of model design, careful attention is paid to provide the required mechanisms with which to incorporate additional and modificatory changes in the disposition of nuclear waste material.

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APPENDIX A - WORK PACKAGE REPORT

Fundamental to the AFR MIS is the AFR Status Report. Once each month this document reports on each work package (e.g., contract) which exists between the DOE and the contractor. The report is paramount to the successful updating of the AFR MIS, for this report relays information between sponsor and contractor. Any discrepancies are quickly identified and resolved.

The AFR Status Report consists of three parts. The first contains general work package information and is shown in Figure 32. The work element number (WEN) is determined by the functional WBS, since one of the main uses of this report is in the tracking of funds. DOE allocates dollar resources via the functional WBS. A description of the scope of work is given along with summary cost data.

Part Two, Figure 33, gives detailed planned and actual cost breakdowns for the associated work package. Projected costs are given in addition to current expenditures.

The interrelationship tracking program and the milestone extraction program act upon the activity data of

Part Three of the AFR Status Report, as shown in Figure 34. For each activity a start date, an end date, a name, and the relationships are listed. Many of the predecessors are from work packages other than this one. This chain of interrelationships permits a computerized tracking of all related activities within the AFR MIS--the basis for the AFR MIS module of System One.

STATUS REPORT SPENT FUEL STORAGE 06/28/80 3- 1
7

WEN NO	PREVIOUS ID	PROGRAM	STATUS
1213.		D F	NEEDED

TITLE: FACILITY IMPLEMENTATION-MORRIS

CONTRACTOR: UNKNOWN

PRINCIPAL INVESTIGATOR:

DESCRIPTION:
THIS TASK WILL PROVIDE FOR:

1) LICENSE FOR OPERATION OF MORRIS OPERATION AS A DOE OWNED AFR SPENT FUEL STORAGE FACILITY WITH A RERACK OF EXISTING POOLS FOR INCREASED STORAGE CAPACITY,
2) CONSTRUCTION OF ADDITIONAL STORAGE POOLS, AND
3) STARTUP AND OPERATION OF MORRIS OPERATION.

START DATE: 01 OCT 80 END DATE: INDEFINITE

DOMESTIC
FUNDING (1000$):

FY79 COST	FY80 COST	FY80 EST
0	0	0

B & R: AS050510
COST ACCT: D1213.

FOREIGN
FUNDING (1000$):

FY79 COST	FY80 COST	FY80 EST
0	0	0

B & R: AS051000
COST ACCT: F1213.

LINE ITEM 79-1-P
FUNDING (1000$):

FY79 COST	FY80 COST	FY80 EST
0	0	0

B & R: 39-AS05051
COST ACCT:

Figure 32. General Information Section of Work Package for GE-Morris

STATUS REPORT SPENT FUEL STORAGE 08/14/80 31- 2
133

1213. FINANCIAL DATA

B&R #: AS050510 FUND TYPE: DOMESTIC
COST ACCOUNT #: D1213.

CURRENT B/O FY EST

FY79	FY80	FY81	FY82	FY83	FY84	FY85	FY86	FY87	FY88
0	0	0	0	6000	4200	5000	4000	0	0

(1980) CURRENT MONTHLY PLAN

OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEPT
0	0	0	0	0	0	0	0	0	0	0	0

B&R #: AS051000 FUND TYPE: FOREIGN
COST ACCOUNT #: F1213.

CURRENT B/O FY EST

FY79	FY80	FY81	FY82	FY83	FY84	FY85	FY86	FY87	FY88
0	0	0	0	0	0	0	0	0	0

B&R #: 39-AS05051 FUND TYPE: LINE ITEM 79-1-P

CURRENT B/O FY EST

FY79	FY80	FY81	FY82	FY83	FY84	FY85	FY86	FY87	FY88
0	0	25000	1000	6200	4500	900	12000	0	0

Figure 33. Financial Section of Work Package for GE-Morris

STATUS REPORT SPENT FUEL STORAGE 06/28/80 3- 2
8

ACTIVITY LIST:

WEN NO: 1213.
START: 01 OCT 80 END: INDEFINITE
TITLE: FACILITY IMPLEMENTATION-MORRIS

ACTIVITY: 1213.1
START: 01 OCT 81 END: 31 JAN 87 % COMPLETE 0
NAME: LICENSING

SUB-ACTY: 1213.1.1
START: 01 OCT 81 END: 02 OCT 81 % COMPLETE 0
NAME: CONSIDER LICENSE APPLICATION ACTIVITIES
AND MAKE APPLICATION FOR TRANSFER AND
RERACK
** PROGRAM CONTROLLED **

PREDECESSORS	SUCCESSORS
1212.1.1	1213.1.2
1211.3	
1121.4	

SUB-ACTY: 1213.1.2
START: 02 OCT 81 END: 30 SEPT 82 % COMPLETE 0
NAME: PREPARE TO RECEIVE LICENSE FOR TRANSFER
AND RERACK
** DELIVERABLE **

PREDECESSORS	SUCCESSORS
1213.1.1	1213.3.1

SUB-ACTY: 1213.1.3
START: 01 JULY 84 END: 01 JULY 84 % COMPLETE 0
NAME: SUBMIT APPLICATION FOR ADD-ON POOL
LICENSE
** PROGRAM CONTROLLED **

PREDECESSORS	SUCCESSORS
1213.1.2	1213.1.4
1212.1.2	
1142.3.6	

SUB-ACTY: 1213.1.4
START: 02 JULY 84 END: 31 JAN 87 % COMPLETE: 0
NAME: PREPARE TO RECEIVE LICENSE FOR ADD-ON
POOL
** CONTRACTOR MILESTONE **

PREDECESSORS	SUCCESSORS
1213.1.3	1213.3.2

ACTIVITY: 1213.2
START: 01 OCT 81 END: 31 JAN 87 % COMPLETE: 0
NAME: CONSTRUCTION DESIGN

SUB-ACTY: 1213.2.1
START: 01 OCT 81 END: 30 JUNE 82 % COMPLETE: 0
NAME: CONSIDER DESIGN PACKAGE AND COMPLETE
RERACK DESIGN
** CONTRACTOR MILESTONE **

PREDECESSORS	SUCCESSORS
1212.2.9	1213.3.1
1414.3	1213.2.2
1423.2.1	

SUB-ACTY: 1213.2.2
START: 01 ACT 84 END: 31 JAN 87 % COMPLETE: 0
NAME: CONDUCT ADD-ON POOL CONSTRUCTION DESIGN

PREDECESSORS	SUCCESSORS
1213.2.1	1213.3.2
1211.3	
1413.2	
1415.3.2	

ACTIVITY: 1213.3
START: 01 OCT 82 END: 26 FEB 89 % COMPLETE: 0
NAME: CONSTRUCTION

Figure 34. Activity Section of Work Package for GE-Morris

SUB-ACTY: 1213.3.1
START: 01 OCT 82 END: 30 JUNE 84 % COMPLETE: 0
NAME: PERFORM RERACK CONSTRUCTION
** CONTRACTOR MILESTONE **

PREDECESSORS	SUCCESSORS
1213.2.1	1213.4.3
1213.1.2	

SUB-ACTY: 1213.3.2
START: 01 FEB 87 END: 26 FEB 89 % COMPLETE: 0
NAME: PERFORM ADD-ON POOL CONSTRUCTION
** CONTRACTOR MILESTONE **

PREDECESSORS	SUCCESSORS
1213.1.4	1213.4.5
1213.2.2	

ACTIVITY: 1213.4
START: 01 JUL 84 END: INDEFINITE % COMPLETE: 0
NAME: OPERATION

SUB-ACTY: 1213.4.1
START: 01 JULY 84 END: 02 JULY 84 % COMPLETE: 0
NAME: START OPERATION WITH RERACK

PREDECESSORS	SUCCESSORS
1213.3.1	1213.4.4

SUB-ACTY: 1213.4.2
START: 03 JULY 84 END: INDEFINITE % COMPLETE: 0
NAME: OPERATE USING RERACK

PREDECESSORS
1213.4.3

COORDINATED
1213.4.2
1213.4.6

SUB-ACTY: 1213.4.3
START: 27 FEB 89 END: 28 FEB 89 % COMPLETE: 0
NAME: RECEIVE FUEL IN ADD-ON POOL

PREDECESSORS	SUCCESSORS
1213.3.2	1213.4.6

SUB-ACTY: 1213.4.4
START: 01 MAR 89 END: INDEFINITE % COMPLETE: 0
NAME: OPERATE FACILITY WITH ADD-ON POOL

PREDECESSORS
1213.4.5

Figure 34. (continued)

APPENDIX B - VERIFICATION ANALYSIS

Numerous models are verified by results of different techniques applied to the problem in question. For example, the mathematical model discussed in reference 17 is verified by a GASP IV simulation model. However, there does not exist any available model, mathematical or otherwise, with which to check the accuracy of the simulation model of the back-end of the nuclear fuel cycle. This fact leads to the exclusive use of hand calculations and temporary output statements in the verification of this model. Hand calculations utilizing the real-case data become prohibitively difficult, since these data contain information on over two hundred individual reactors and numerous storage facilities. Therefore, abbreviated data from the various input sources are employed in the verification analysis.

Table 13 lists the information on the two AFR facilities and the one repository that are used in the verification. The foreign spent fuel discharge amounts are given in Table 14. The final table in the section, Table 15, depicts the quantities of spent fuel discharged from domestic reactors. Notice that only eighteen

Table 13. Storage Site Input Data for Verification Analysis

STORAGE SITE	ON-LINE TIME	CAPACITY (mtu)	HANDLING RATE (mtu/year)	AMOUNT OF EXISTING SPENT FUEL (mtu)	LATITUDE	LONGITUDE
AFR FACILITY NUMBER ONE	80.5	1750	750	0	33.2	81.4
AFR FACILITY NUMBER TWO	82.5	1100	250	350	41.4	88.4
REPOSITORY NUMBER ONE	83.6	41000	1800	0	47.0	118.0

Table 14. Foreign Input Data for Verification Analysis

PSEUDO-REACTOR	LATITUDE	LONGITUDE	SPENT FUEL DISCHARGE AMOUNT (mtu)				
			1979	1980	1981	1982	1983+
300	33.8	118.2	0	0	18	16	0
301	40.7	73.5	0	0	16	16	0
302	32.8	79.9	0	0	16	18	0

Table 15. Domestic Reactor Input Data for Verification Anaylsis

REACTOR NUMBER	LATITUDE	LONGITUDE	FULL-CORE RESERVE CAPACITY	SPENT FUEL DISCHARGE AMOUNT (mtu)									
				79	80	81	82	83	84	85	86	87	88
1	31.2	85.1	70.8	10	20	45	45	45	0	0	0	0	0
2	31.2	85.1	70.8	32	55	10	61	75	32	30	0	0	0
5	33.4	112.9	102.9	14	32	32	24	24	24	24	45	45	45
6	33.4	112.9	102.9	0	0	0	0	0	0	0	0	0	0
7	33.4	112.9	102.9	45	46	67	89	89	89	23	23	23	23
8	35.3	93.2	82.1	20	30	23	12	12	45	28	21	21	21
9	35.3	93.2	73.5	67	67	45	55	55	55	60	60	60	60
10	38.4	76.4	82.9	0	0	0	0	0	0	0	0	0	0
11	38.4	76.4	84.4	23	23	23	34	34	34	34	34	20	20
12	42.0	70.6	112.5	40	40	30	30	70	70	70	70	70	70
13	42.0	70.6	89.4	20	10	10	10	10	10	10	10	10	10
14	34.4	80.2	70.5	0	14	20	20	20	20	20	20	20	20
15	33.0	78.0	109.2	0	30	30	9	27	27	27	27	27	27
16	33.0	78.0	109.2	0	0	26	27	27	27	27	27	27	27
17	35.6	78.9	72.5	60	60	9	9	9	5	5	10	20	20
18	35.6	78.9	72.5	44	44	38	38	38	49	49	49	49	21
19	35.6	78.9	72.5	0	30	30	30	0	0	0	0	0	0
20	35.6	78.9	72.5	10	0	20	0	20	20	20	50	55	55

reactors are employed. Since the end of the simulation is selected as time 87.9, the discharge data for the year 2000 to the year 2020 are not required. The accident analysis module is not utilized in this process. All other options are allowed, including storage against FCR and in the temporary file. The shipment priority is nearness for discharges to both the AFR facilities and the repository.

After the aforementioned input data are accumulated, the simulation is performed. The output is shown in the multiple pages of Figure 35. The complete verification process involves examining the output and input from many different directions. For illustrative purposes in this appendix, the discharges from Reactor Two are highlighted and followed.

At time 79.3, this reactor discharges 32 metric tons of uranimum (mtu) and stores the fuel against its FCR capacity of 70.8 mtu. Both these figures agree with the input data given in Table 15; hence, the GASP IV input subroutine, which gathers the data from System One and places said data in the event file, is working properly. The following year, time 80.3, finds that 55 mtu is discharged from Reactor Two. Since there is not enough at-reactor storage space (only 38.8 mtu remain) to hold an

```
REACTOR     1.  STORED    10.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    10.0 MTU; FCR IS    70.8 MTU
REACTOR     2   STORED    32.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    32.0 MTU; FCR IS    70.8 MTU
REACTOR     5.  STORED    14.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    14.0 MTU; FCR IS   102.9 MTU
REACTOR     7.  STORED    45.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    45.0 MTU; FCR IS   102.9 MTU
REACTOR     8.  STORED    20.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    20.0 MTU; FCR IS    82.1 MTU
REACTOR     9.  STORED    67.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    67.0 MTU; FCR IS    73.5 MTU
REACTOR    11.  STORED    23.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    23.0 MTU; FCR IS    84.4 MTU
REACTOR    12.  STORED    40.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    40.0 MTU; FCR IS   112.5 MTU
REACTOR    13.  STORED    20.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    20.0 MTU; FCR IS    89.4 MTU
REACTOR    17.  STORED    60.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    60.0 MTU; FCR IS    72.5 MTU
REACTOR    18.  STORED    44.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    44.0 MTU; FCR IS    72.5 MTU
REACTOR    20.  STORED    10.0 MTU AGAINST FCR AT TIME    79.3:  TOTAL AGAINST FCR    10.0 MTU; FCR IS    72.5 MTU
REACTOR     1.  STORED    20.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    30.0 MTU; FCR IS    70.8 MTU
REACTOR NUMBER    2   CANNOT STORE  55. MTU AT  80.3
REACTOR     5.  STORED    32.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    46.0 MTU; FCR IS   102.9 MTU
REACTOR     7.  STORED    46.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    91.0 MTU; FCR IS   102.9 MTU
REACTOR     8.  STORED    30.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    50.0 MTU; FCR IS    82.1 MTU
REACTOR NUMBER    9.  CANNOT STORE  67. MTU AT  80.3
REACTOR    11.  STORED    23.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    46.0 MTU; FCR IS    84.4 MTU
REACTOR    12.  STORED    40.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    80.0 MTU; FCR IS   112.5 MTU
REACTOR    13.  STORED    10.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    30.0 MTU; FCR IS    89.4 MTU
REACTOR    14.  STORED    14.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    14.0 MTU; FCR IS    70.5 MTU
REACTOR    15.  STORED    30.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    30.0 MTU; FCR IS   109.2 MTU
REACTOR NUMBER   17.  CANNOT STORE  60. MTU AT  80.3
REACTOR NUMBER   18.  CANNOT STORE  44. MTU AT  80.3
REACTOR    19.  STORED    30.0 MTU AGAINST FCR AT TIME    80.3:  TOTAL AGAINST FCR    30.0 MTU; FCR IS    72.5 MTU

FACILITY NUMBER  1 HAS INCREASED CAPACITY BY 1750. MTU AT TIME=  80.5
   MAX CAPACITY IS    1750.0; HANDLING RATE IS    750.0 MTU PER YEAR
REACTOR     9  STORED   67.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR     7  STORED   91.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    17  STORED   60.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    12  STORED   80.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR     8  STORED   50.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    18  STORED   44.9 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    11  STORED   46.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR     2  STORED   32.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR     5  STORED   46.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR     1  STORED   30.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    19  STORED   30.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    13  STORED   30.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    15  STORED   30.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    14  STORED   14.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    20  STORED   10.0 MTU AT TIME    80.5 IN FACILITY    1 TO REGAIN FCR
REACTOR    18 HAS REMOVED     44.0 MTU FROM TEMPORARY STORAGE
         FUEL STORED IN FACILITY NUMBER     1 AT TIME    80.5
REACTOR     2 HAS REMOVED     46.0 MTU FROM TEMPORARY STORAGE AT TIME     81.
         THIS AMOUNT WAS DICTATED BY THE HANDLING RATE OF FACILITY    1
```

Figure 35. Output Listing from Verification Analysis Input Data

```
FACILITY NUMBER   1 HAS EXCEEDED ITS HANDLING RATE AT TIME    81.3
REACTOR NUMBER  300. CANNOT STORE  18. MTU AT    81.3

FACILITY NUMBER   1 HAS EXCEEDED ITS HANDLING RATE AT TIME    81.3
REACTOR NUMBER  301. CANNOT STORE  16. MTU AT    81.3

FACILITY NUMBER   1 HAS EXCEEDED ITS HANDLING RATE AT TIME    81.3
REACTOR NUMBER  302. CANNOT STORE  16. MTU AT    81.3

FACILITY NUMBER   1 HAS EXCEEDED ITS HANDLING RATE AT TIME    81.3
REACTOR      1.  STORED   45.0 MTU AGAINST FCR AT TIME   81.3:  TOTAL AGAINST FCR   45.0 MTU; FCR IS   70.8 MTU

FACILITY NUMBER   1 HAS EXCEEDED ITS HANDLING RATE AT TIME    81.3
REACTOR      2   STORED   10.0 MTU AGAINST FCR AT TIME   81.3:  TOTAL AGAINST FCR   10.0 MTU; FCR IS   70.8 MTU

FACILITY NUMBER   1 HAS EXCEEDED ITS HANDLING RATE AT TIME    81.3
REACTOR      5.  STORED   32.0 MTU AGAINST FCR AT TIME   81.3:  TOTAL AGAINST FCR   32.0 MTU; FCR IS  102.9 MTU

FACILITY NUMBER   1 HAS EXCEEDED ITS HANDLING RATE AT TIME    81.3
REACTOR      7.  STORED   67.0 MTU AGAINST FOR AT TIME   81.3:  TOTAL AGAINST FCR   67.0 MTU; FCR IS  102.9 MTU
*****REACTOR     300. STORED   16.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR     301. STORED   16.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR     302. STORED   18.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR       1. STORED   45.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR       2  STORED   61.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR       5. STORED   24.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR       7. STORED   89.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR       8. STORED   12.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR       9. STORED   55.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR      11. STORED   34.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR      12. STORED   30.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR      13. STORED   10.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR      14. STORED   20.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR      15. STORED    9.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR      16. STORED   27.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR      17. STORED    9.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR      18. STORED   38.0 MTU IN FACILITY   1 AT TIME  82.3
*****REACTOR      19. STORED   30.0 MTU IN FACILITY   1 AT TIME  82.3

FACILITY NUMBER   2 HAS INCREASED CAPACITY BY  1100. MTU AT TIME=  82.5
    MAX CAPACITY IS     1100.0; HANDLING RATE IS     250.0 MTU PER YEAR

REACTOR      7 STORED   67.0 MTU AT TIME   82.5 IN FACILITY   2 TO REGAIN FCR
REACTOR      1 STORED   45.0 MTU AT TIME   82.5 IN FACILITY   2 TO REGAIN FCR
REACTOR      9 STORED   45.0 MTU AT TIME   82.5 IN FACILITY   2 TO REGAIN FCR
REACTOR     18 STORED   38.0 MTU AT TIME   82.5 IN FACILITY   2 TO REGAIN FCR
REACTOR     19 STORED   30.0 MTU AT TIME   82.5 IN FACILITY   2 TO REGAIN FCR
REACTOR      5 REMOVED   25.0 MTU FROM ITS AR POOL AT TIME   82.5
         THIS AMOUNT WAS DICTATED BY THE HANDLING RATE OF FACILITY   2
         THIS REACTOR STILL IS UTILIZING    6.3 PERCENT OF ITS FCR CAPACITY
FOLLOWING STATISTICS AT TIME   82.5


                                          **STATISTICS FOR TIME-PERSISTENT VARIABLES**
              MEAN          STD DEV        MINIMUM        MAXIMUM       TIME INTERVAL   CUR. VALUE

FACIL1     0.4133E+00     0.1085E+00     0.0            0.7389E+00     0.2000E+01      0.7389E+00

                                          **STATISTICS FOR TIME-PERSISTENT VARIABLES**
              MEAN          STD DEV        MINIMUM        MAXIMUM       TIME INTERVAL   CUR. VALUE

FACIL2                                          NO VALUES RECORDED
```

Figure 35. (continued)

```
*****REACTOR     1.  STORED   45.0 MTU IN FACILITY  1 AT TIME   83.3
*****REACTOR     2   STORED   75.0 MTU IN FACILITY  1 AT TIME   83.3

FACILITY NUMBER   2 HAS EXCEEDED ITS HANDLING RATE AT TIME   83.3
*****REACTOR     5.  STORED   24.0 MTU IN FACILITY  1 AT TIME   83.3

FACILITY NUMBER   2 HAS EXCEEDED ITS HANDLING RATE AT TIME   83.3
*****REACTOR     7.  STORED   89.0 MTU IN FACILITY  1 AT TIME   83.3

FACILITY NUMBER   2 HAS EXCEEDED ITS HANDLING RATE AT TIME   83.3
*****REACTOR     8.  STORED   12.0 MTU IN FACILITY  1 AT TIME   83.3

FACILITY NUMBER   2 HAS EXCEEDED ITS HANDLING RATE AT TIME   83.3
*****REACTOR     9.  STORED   55.0 MTU IN FACILITY  1 AT TIME   83.3
*****REACTOR    11.  STORED   34.0 MTU IN FACILITY  1 AT TIME   83.3
*****REACTOR    12.  STORED   70.0 MTU IN FACILITY  1 AT TIME   83.3
*****REACTOR    13.  STORED   10.0 MTU IN FACILITY  1 AT TIME   83.3
REPOSITORY NUMBER  1 HAS INCREASED CAPACITY BY 41000.0 MTU AT TIME=  83.6
    MAX CAPACITY IS  41000.0; HANDLING RATE IS  1800.0 MTU PER YEAR

REACTOR     20 STORED   40.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR     18 STORED   38.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR     15 STORED   57.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR     16 STORED   53.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR     14 STORED   20.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR      8 STORED   23.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR     11 STORED   23.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR     12 STORED   30.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR      2 STORED   10.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR     17 STORED    9.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR     13 STORED   10.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR      5 STORED    7.0 MTU AT TIME    83.6 IN REPOSITORY    1 TO REGAIN FCR
REACTOR      2 HAS REMOVED      9.0 MTU FROM TEMPORARY STORAGE
         FUEL STORED IN REPOSITORY NUMBER    1  AT TIME    83.6

REACTOR    301 HAS REMOVED     16.0 MTU FROM TEMPORARY STORAGE
         FUEL STORED IN REPOSITORY NUMBER    1  AT TIME    83.6

REACTOR    302 HAS REMOVED     16.0 MTU FROM TEMPORARY STORAGE
         FUEL STORED IN REPOSITORY NUMBER    1  AT TIME    83.6

REACTOR    300 HAS REMOVED     18.0 MTU FROM TEMPORARY STORAGE
         FUEL STORED IN REPOSITORY NUMBER    1  AT TIME    83.6

REACTOR     17  HAS REMOVED    60.0 MTU FROM TEMPORARY STORAGE
         FUEL STORED IN REPOSITORY NUMBER    1  AT TIME    83.6

REACTOR      9  HAS REMOVED    67.0 MTU FROM TEMPORARY STORAGE
         FUEL STORED IN REPOSITORY NUMBER    1  AT TIME    83.6

FACILITY    1  REMOVED  307.0 MTU AND PLACED FUEL INTO REPOSITORY    1  AT TIME    83.7
        AMOUNT WAS LIMITED BY FACILITY HANDLING RATE

FACILITY    2  REMOVED  250.0 MTU AND PLACED FUEL INTO REPOSITORY    1  AT TIME    83.7
        AMOUNT WAS LIMITED BY FACILITY HANDLING RATE

*****REACTOR     2   STORED   32.0 MTU IN REPOSITORY 1  AT TIME   84.3
*****REACTOR     5.  STORED   24.0 MTU IN REPOSITORY 1  AT TIME   84.3
*****REACTOR     7.  STORED   89.0 MTU IN REPOSITORY 1  AT TIME   84.3
*****REACTOR     8.  STORED   45.0 MTU IN REPOSITORY 1  AT TIME   84.3
*****REACTOR     9.  STORED   55.0 MTU IN REPOSITORY 1  AT TIME   84.3
*****REACTOR    11.  STORED   34.0 MTU IN REPOSITORY 1  AT TIME   84.3
*****REACTOR    12.  STORED   70.0 MTU IN REPOSITORY 1  AT TIME   84.3
*****REACTOR    13.  STORED   10.0 MTU IN REPOSITORY 1  AT TIME   84.3
*****REACTOR    14.  STORED   20.0 MTU IN REPOSITORY 1  AT TIME   84.3
*****REACTOR    15.  STORED   27.0 MTU IN REPOSITORY 1  AT TIME   84.3
```

Figure 35. (continued)

```
FACILITY     1 REMOVED  750.0 MTU AND PLACED FUEL INTO REPOSITORY  1  AT TIME   84.7
        AMOUNT WAS LIMITED BY FACILITY HANDLING RATE

*****REACTOR      2   STORED    30.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR      5.  STORED    24.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR      7.  STORED    23.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR      8.  STORED    28.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR      9.  STORED    60.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR     11.  STORED    34.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR     12.  STORED    70.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR     13.  STORED    10.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR     14.  STORED    20.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR     15.  STORED    27.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR     16.  STORED    27.0 MTU IN REPOSITORY  1  AT TIME    85.3
*****REACTOR     17.  STORED     5.0 MTU IN REPOSITORY  1  AT TIME    85.3
FACILITY     2 REMOVED  250.0 MTU AND PLACED FUEL INTO REPOSITORY  1  AT TIME  85.7
        AMOUNT WAS LIMITED BY FACILITY HANDLING RATE

*****REACTOR      5.  STORED    45.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR      7.  STORED    23.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR      8.  STORED    21.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR      9.  STORED    60.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR     11.  STORED    34.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR     12.  STORED    70.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR     13.  STORED    10.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR     14.  STORED    20.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR     15.  STORED    27.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR     16.  STORED    27.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR     17.  STORED    10.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR     18.  STORED    49.0 MTU IN REPOSITORY  1  AT TIME    86.3
*****REACTOR     20.  STORED    50.0 MTU IN REPOSITORY  1  AT TIME    86.3

                                **STATISTICS FOR TIME-PERSISTENT VARIABLES**
            MEAN          STD DEV       MINIMUM      MAXIMUM      TIME INTERVAL  CUR. VALUE

FACIL1     0.4543E+00    0.2865E+00    0.0          0.9920E+00    0.7500E+01     0.0

                                **STATISTICS FOR TIME-PERSISTENT VARIABLES**
            MEAN          STD DEV       MINIMUM      MAXIMUM      TIME INTERVAL  CUR. VALUE

FACIL2     0.2678E+00    0.1818E+00    0.0          0.5455E+00    0.5500E+01     0.0

                                **STATISTICS FOR TIME-PERSISTENT VARIABLES**
            MEAN          STD DEV       MINIMUM      MAXIMUM      TIME INTERVAL  CUR. VALUE

REPOS1     0.7031E-01    0.2984E-01    0.0          0.1139E+00    0.4400E+01     0.1139E+00

FACILITY NUMBER  1 HAD  46 SHIPMENTS

FACILITY NUMBER  2 HAD   6 SHIPMENTS

REPOSITORY NUMBER  1 HAD  72 SHIPMENTS
```

Figure 35. (continued)

additional 55 mtu, Reactor Two cannot store the fuel. This amount is tracked within the temporary file.

AFR Facility One begins operation at time 80.5. Immediately thereafter, fuel is transferred from the FCR sections of the various reactors. After all reactors regain their FCR capacities, fuel tracked within the temporary file is removed and stored in Facility One. Reactor Two can only transfer 46 mtu out of the 55 mtu held in the temporary file. This amount is limited by the handling rate of the AFR facility. An addition of the amount of fuel shipped to Facility One confirms that the handling rate is maximized.

For the next year, Facility One cannot accept any fuel. Reactor Two stores fuel against FCR at time 81.3. The total against FCR is 10 mtu, verifying that the file containing the amounts of fuel stored in the FCR section is reset to zero when the fuel is transferred to Facility One at time 80.5. The next discharge from Reactor Two is shipped to Facility One, since the handling rate for this facility is reset at time 81.5.

Reactor Two continues to store fuel in the AFR facilities. The next check point occurs at time 83.6. Repository One comes on-line, reactors regain their FCR capacities, and then fuel is transferred from the temporary file. At this time, 9 mtu from Reactor Two is removed from the file. This amount, when added to the 46 mtu removed at time 80.5, equals the total amount of fuel (55 mtu) placed into the temporary file at time 80.3. Discharges from Reactor Two are directed to the repository until time 86.3, the end-of-simulation.

Comprehensive verification requires the examination of numerous reactors and facilities by the method just described. For example, another check happens at time 85.7. Facility Two removes 250 mtu of fuel and places the fuel into Repository One. This figure agrees with the handling rate of Facility Two as given in the input data. The final statistics, given at the end of the output, confirm that both AFR facilities accept fuel and then transfer the fuel to the repository.

The comparison between input and output continues until all facets of model operation are verified. In addition, many of the subroutines, as they are coded, are independently operated and checked. These steps confirm the exactness of this descriptive simulation model.

APPENDIX C - EVENT CODE DESCRIPTION

GASP IV controls the logic of System Two of the simulation model by keying on the event code associated with each entry. An example of an entry is reactor number thirty discharging 43 mtu of spent fuel at time 87.3. Along with this information, the user attaches an event code of one. GASP IV reads the event code and directs the simulation.

Table 16 describes each of the six event codes employed in the model. A specific requirement of GASP IV is that the event code exists as attribute two. Also, the ranking attribute of the model (time in this simulation) is stored as attribute one. All other attributes are assigned at the discretion of the user. GASP IV provides for a maximum of twenty-five attributes per entry.

Table 16. Description of Event Code and Attributes

EVENT CODE	ATTRIBUTE NUMBER AND DESCRIPTION								
	1	2	3	4	5	6	7	8	9
1(STORAGE DEMAND) 1	TIME OF SPENT FUEL DISCHARGE	EVENT CODE	AMOUNT OF DISCHARGE (mtu)	REACTOR NUMBER	UTILITY CODE	STATE CODE	NERC CODE	LATITUDE	LONGITUDE
2(AFR STORAGE SUPPLY) 2	TIME OF OPERATION	EVENT CODE	CAPACITY (mtu)	AFR FACILITY NUMBER	HANDLING RATE (mtu/year)	AMOUNT OF ANY EXISTING FUEL (mtu)	N/A	N/A	N/A
3(STATISTICS REPORTING) 3	TIME OF REPORTING EVERY YEAR	EVENT CODE	N/A	N/A	N/A	N/A	N/A	N/A	N/A
4(COLLECT PLOT DATA) 4	TIME OF COLLECTING EVERY TWO-TENTHS OF A YEAR	EVENT CODE	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5(REPOSITORY STORAGE SUPPLY) 5	TIME OF OPERATION	EVENT CODE	CAPACITY (mtu)	REPOSITORY NUMBER	HANDLING RATE (mtu/year)	AMOUNT OF ANY EXISTING FUEL (mtu)	N/A	N/A	N/A
6(AFR TO REPOSITORY TRANSFER) 6	TIME OF TRANSFER-EVERY YEAR	EVENT CODE	N/A	N/A	N/A	N/A	N/A	N/A	N/A

APPENDIX D - SPENT FUEL SURVEY FORM

Precision of the individual reactor discharge data is mandatory for reliable model results. With approximately sixty utilities responsible for nearly two hundred operating and proposed reactors, the task of ensuring such precision is formidable. A prototype questionnaire is designed to assist in transmitting data between the utilities and the Stoller Database. A separate survey form is completed for each reactor. As a questionnaire is produced, an auxiliary program reads the Stoller Database and inserts data where appropriate. The resultant document is then sent to the required utility. An individual just corrects the form; relieving him of the time-consuming task of gathering the information and filling out the survey from scratch.

Figure 36 presents a complete survey form for the Farley-1 reactor. The form requests more information than presently contained within the Stoller Database; hence, some fields are blank. These additional facts are required for future endeavors.

The S. M. Stoller Corporation adapted the format and production methodology designed into the prototype survey form for the actual questionnaire. The first mailing occurred during the summer of 1980. The Stoller Database is expected to be periodically updated via the survey form.

DOMESTIC SPENT FUEL STORAGE SURVEY

DATE: ____________

Utility Name: ALABAMA POWER CO
Reactor Name: FARLEY-1
Location (Latitude, Longtitude): 31 Degrees,
~~35 Degrees,~~ 6 Minutes

Person(s) Supplying Data: ____________
Mailing Address: ____________

Telephone Number: ____________

REACTOR

Type: Pressurized Water XX Boiling Water ____
Vendor: W
Net Capacity: MWe 829. MWt ____
Commercial Operations Date (Month and Year) 1973
If not operational, state:
Licensing Status: ____
Construction Status (Month and Year): ____
Limited Work Authorization (Construction Permit): ____
Completion (or Forecast): ____
Full Core Size: Number of Assemblies 157
Assembly Weight MTU 0.451

POOL CHARACTERISTICS

Number of Rectangular Spent Fuel Pools: 1

Length (ft) 37.9
Width (ft) 21.6
Depth (ft) 39.0
Center-to-Center Spacing (in.) 10.37500

Figure 36. Domestic Spent Fuel Storage Survey

Number of "L" Shaped Fuel Pools: ________

A (ft) ______________
B (ft) ______________
C (ft) ______________
D (ft) ______________
Depth (ft) ______________
Center-to-Center Spacing (in.) ________

Note Any Variations in Depth

Number of Other Pools That Potentially Could be Used to Store Spent Fuel: ________

Length (ft) ______________
Width (ft) ______________
Depth (ft) ______________
Potential Center-to-Center Spacing if Known (in.) ______

Existing Storage Capacity (Fuel Assemblies): 675 (MTU) 304.425

Licensed Storage Capacity (Fuel Assemblies): ______ (MTU) ________

Is Pool Now Shared or Forecasted to be Shared by Other Reactor(s):
Yes ____ No XX

For Shared Pool (Before May 1, 1979)

Names of All Reactors Sharing Pool	Fuel Stored (By React	

Cask Handling Area (ft. sq.), if Handled in Storage Pool: ________

Figure 36. (continued)

Cask Handling Capability:

Cask Handling Limitation
(i.e., Crane Capacity,
Pool Depth)

Spent Fuel Shipping Cask Ownership:

Cask Type	Number	Type of Ownership (Leased or Purchase	Purchase

Assemblies):

Increased Spent Fuel Pool Storage Capacity

Has Action Been Taken to Increase Spent Fuel Pool Storage Capacity?
No ______ Yes XX

If Yes, Describe Action Taken and Date:

Future Plans for Increased Storage Capacity:

Method of Increasing Storage Capacity*	Year	MTU	Total Storage Capacity After Increase	Center to Center	Licensing

*i.e., Rerack; Double Tier; Pool Modification

Figure 36. (continued)

Total Storage Capacity After Maximum Possible Expansion:

Fuel Assemblies 675 MTU 304.425

Institutional impediments (i.e., Statutory limits on Expansions)

SPENT FUEL STORAGE PROGRAM

As of May 1, 1979.
Total Amount of Spent Fuel Discharged: Assemblies 0 MTU UUUUUUUUUU
Amt. of Spent Fuel in Storage at Reactor Site: Assemblies 0 MTU 0.000
If There is any difference between the above two responses, please explain:

Age and Origin of the Stored Fuel:

No. of Assemblies	MTU	Age of Fuel in Years As of May 1, 1979	Number of Leakers or Defective Fuel Assemblies	Origin
0	0.000		0	

Projected Spent Fuel Discharges:

Month	Year	Bases for Calculation: Thermal Power MWt	Capacity Factor %	Average Burnup of Discharged	Number of	
	1979					
	1980					
	1981					
	1982					
	1983					
	1984					
	1985					
	1986					
	1987					
	1988					
	1989					
	1990					

Figure 36. (continued)

SPENT FUEL SHIPPING PROGRAM

Capability: Rail ____________ Truck ____________ Barge ____________

Institutional Impediments: ____________

Shipments Before April 1, 1979:
Number of Assemblies: ____________ MTU ____________
Location: ____________

Plans for Future Shipments Off Site After April 1, 1979:

Year	Number of Assemblies	MTU	Mode of Transportation	____________	Cask to
1979					
1980					
1981					
1982					
1983					
1984					
1985					
1986					
1987					
1988					
1989					
1990					

FULL CORE RESERVE

Is Full Core Reserve Intended to be Maintained for Future Reactor Operation?

Yes ____ No ____

State Reason ____________

Figure 36. (continued)

OPERATING HISTORY OF REACTOR

Cycle #	Refueling Shutdown Date Mo.	Year	Power MWt	Cap. Factor %	Burnup *	Assem. Disc. *	Failed		Effective Full Power

*A batch or region is defined as a group charged to the reactor at the same time and discharged at the same time.

Figure 36. (continued)

APPENDIX E - REACTOR NUMBER VERSUS NAME

Programming chores and execution times are reduced by the use of numbers rather than character strings. However, on output reports the term "reactor 35" does not relay much information. The associated literal name, Byron-2, conveys a sense of exactness. Any program can be modified to convert between the different terms; the more conversion, the more execution time. Many of the DOE managers who utilize the model are familiar with the reactor numbers and associated names; thus, at this time, the conversion is not computerized.

Table 17 lists reactor number, reactor name, and affiliated utility. The list includes all operating and proposed reactors which are maintained within the Stoller Database.

Table 17. Cross Reference between Reactor Number and Name

REACTOR I.D.	NAME	UTILITY	REACTOR I.D.	NAME	UTILITY
1	FARLEY-1	ALABAMA POWER CO.	60	CHEROKEE-2	DUKE POWER COMPANY
2	FARLEY-2	ALABAMA POWER CO.	61	CHEROKEE-3	DUKE POWER COMPANY
5	PALO-VERDE-1	ARIZONA PUB SERV CO.	63	BEAVER VALLEY-1	DUQUESNE LIGHT CO
6	PALO-VERDE-2	ARIZONA PUB SERV CO	64	BEAVER VALLEY-2	DUQUESNE LIGHT CO
7	PALO-VERDE-3	ARIZONA PUB SERV CO	65	CRYSTAL RIVER-3	FLORIDA POWER CORP
8	ARKANSAS NUCL ONE-L	ARKANSAS P AND L CO	66	TURKEY POINT-3	FLORIDA P AND L CO
9	ARKANSAS NUCL ONE-2	ARKANSAS P AND L CO	67	TURKEY POINT-4	FLORIDA P AND L CO
10	CALVERT CLIFFS-1	BALTIMORE G AND E CO	68	ST. LUCIE-1	FLORIDA P AND L CO
11	CALVERT CLIFFS-2	BALTIMORE G AND E CO	69	ST. LUCIE-2	FLORIDA P AND L CO
12	PILGRIM-1	BOSTON EDISON CO	70	HATCH-1	GEORGIA POWER CO
13	PILBRIM-2	BOSTON EDISON CO	71	HATCH-2	GEORGIA POWER CO
14	ROBINSON-2	CAROLINA P AND L CO	72	VOGTLE-1	GEORGIA POWER CO
15	BRUNSWICK-2	CAROLINA P AND L CO	73	VOGTLE-2	GEORGIA POWER CO
16	BRUNSWICK-1	CAROLINA P AND L CO	74	RIVER BEND-1	GULF STATES UTLTS CO
17	HARRIS-1	CAROLINA P AND L CO	75	ROVER BEND-2	GULF STATES UTLTS CO
18	HARRIS-4	CAROLINA P AND L CO	78	ALLENS CREEK	HOUSTON L AND P CO
19	HARRIS-2	CAROLINA P AND L CO	79	CLINTON-1	ILLINOIS POWER CO
20	HARRIS-3	CAROLINA P AND L CO	80	CLINTON-2	ILLINOIS POWER CO
22	ZIMMER-1	CINCINNATI G AND E	81	D C COOK-1	IND AND MI ELEC CO
23	PERRY-1	CLEVELAND ELECTRIC	82	D C COOK-2	IND AND MI ELEC CO
24	PERRY-2	CLEVELAND ELECTRIC	83	DUANE ARNOLD	IOWA E L AND P CO
25	DRESDEN-1	COMMONWEALTH EDISON	85	OYSTER CREEK	JERSEY CENTRAL PL CO
26	DRESDEN-2	COMMONWEALTH EDISON	86	FORKED RIVER	JERSEY CENTRAL PL CO
27	DRESDEN-3	COMMONWEALTH EDISON	87	WOLF CREEK	KANSAS G AND E CO
28	QUAD CITIES-1	COMMONWEALTH EDISON	88	SHOREHAM	LONG ISLAND LIGHTING
29	QUAD CITIES-2	COMMONWEALTH EDISON	89	JAMESPORT-1	LONG ISLAND LIGHTING
30	ZION-1	COMMONWEALTH EDISON	90	JAMESPORT-2	LONG ISLAND LIGHTING
31	ZION-2	COMMONWEALTH EDISON	93	WATERFORD-3	LOUISIANA P AND L CO
32	LA SALLE-1	COMMONWEALTH EDISON	94	MAINE YANKEE	MAINE YANKEE AT PWR
33	LA SALLE-2	COMMONWEALTH EDISON	95	THREE MILE ISLAND-1	METROPOLITAN EDISON
34	BYRON-1	COMMONWEALTH EDISON	96	THREE MILE ISLAND-2	METROPOLITAN EDISON
35	BYRON-2	COMMONWEALTH EDISON	97	GRAND GULF-1	MISSISSIPPI P AND L
36	BRAIDWOOD-1	COMMONWEALTH EDISON	98	GRAND GULF-2	MISSISSIPPI P AND L
37	BRAIDWOOD-2	COMMONWEALTH EDISON	99	COOPER	NEBRASKA PUB PWR DIS
38	CONNECTICUT YANKEE	CONN. YANKEE ATOMIC	100	NEP-1	NEW ENGLAND POWER CO
40	INDIAN POINT-2	CONSOLIDATED EDISON	101	NEP-2	NEW ENGLAND POWER CO
41	BIG ROCK POINT	CONSUMERS POWER CO	102	NEW HAVEN-1	N Y STATE E AND G
42	PALISADES	CONSUMERS POWER CO	103	NEW HAVEN-2	N Y STATE E AND G
43	MIDLAND-2	CONSUMERS POWER CO	104	NINE MILE POINT-1	NIAGARA MOHAWK POWER
44	MIDLAND-1	CONSUMERS POWER CO	105	NINE MILE POINT-2	NIAGARA MOHAWK POWER
45	LA CROSSE	DAIRYLAND POWER CORP	106	MILLSTONE-1	NORTHEAST UTILITIES
46	FERMI-2	DETROIT EDISON CO	107	MILLSTONE-2	NORTHEAST UTILITIES
47	GREENWOOD-2	DETROIT EDISON CO	108	MILLSTONE-3	NORTHEAST UTILITIES
48	GREENWOOD-3	DETROIT EDISON CO	109	MONTAGUE-1	NORTHEAST UTILITIES
49	OCONEE-1	DUKE POWER COMPANY	110	MONTAGUE-2	NORTHEAST UTILITIES
50	OCONEE-2	DUKE POWER COMPANY	111	BAILLY	NORTH INDIANA P S CO
51	OCONEE-3	DUKE POWER COMPANY	112	MONTICELLO	NORTHERN STATES PWR
52	MCGUIRE-1	DUKE POWER COMPANY	113	PRAIRIE ISLAND-1	NORTHERN STATES PWR
53	MCGUIRE-2	DUKE POWER COMPANY	114	PRAIRIE ISLAND-2	NORTHERN STATES PWR
54	CATAWBA-1	DUKE POWER COMPANY	116	ERIE-1	OHIO EDISON COMPANY
55	CATAWBA-2	DUKE POWER COMPANY	117	ERIE-2	OHIO EDISON COMPANY
56	PERKINS-1	DUKE POWER COMPANY	118	FORT CALHOUN-1	OMAHA PUBL PWR DIST
57	PERKINS-2	DUKE POWER COMPANY	119	HUMBOLDT BAY	PACIFIC G AND E CO
58	PERKINS-3	DUKE POWER COMPANY	120	DIABLO CANYON-2	PACIFIC G AND E CO
59	CHEROKEE-1	DUKE POWER COMPANY	121	DIABLO CANYON-1	PACIFIC G AND E CO

Table 17. (continued)

REACTOR I.D.	NAME	UTILITY
122	SUSQUEHANNA-1	PENNSYLVANIA P AND L
123	SUSQUEHANNA-2	PENNSYLVANIA P AND L
124	PEACH BOTTOM-2	PHILADELPHIA ELEC CO
125	PEACH BOTTOM-3	PHILADELPHIA ELEC CO
126	LIMERICK-1	PHILADELPHIA ELEC CO
127	LIMERICK-2	PHILADELPHIA ELEC CO
128	TROJAN	PORTLAND GEN ELEC CO
129	PEBBLE SPRINGS-1	PORTLAND GEN ELEC CO
130	PEBBLE SPRINGS-2	PORTLAND GEN ELEC CO
133	FITZPATRICK	POW, AUTH, STATE OF NY
134	INDIAN POINT-3	POW, AUTH, STATE OF NY
137	MARBLE HILL-1	PUB SERVICE INDIANA
138	MARBLE HILL-2	PUB SERVICE INDIANA
139	SEABROOK-1	PSC OF NEW HAMPHSIRE
140	SEABROOK-2	PSC OF NEW HAMPSHIRE
141	BLACK FOX-1	PUB SERV CO OKLAHOMA
142	BLACK FOX-2	PUB SERV CO OKLAHOMA
143	SALEM-1	PUB SERV E AND G CO
144	SALEM-2	PUB SERV E AND G CO
145	HOPE CREEK-1	PUB SERV E AND E CO
146	HOPE CREEK-2	PUB SERV E AND E CO
151	SKAGIT-1	PUGET SOUND P AND L
152	SKAGIT-2	PUGET SOUND P AND L
153	GINNA	ROCHESTER G AND E CO
154	STERLING	ROCHESTER G AND E CO
155	RANCHO SECO-1	SACREMENTO MUN UT DI
158	SUMMER	SO CAROLINA E AND G
159	SAN ONOFRE-1	SO CALIF EDISON CO
160	SAN ONOFRE-2	SO CALIF EDISON CO
161	SAN ONOFRE-3	SO CALIF EDISON CO
162	SOUTH TEXAS PLANT-1	HOUSTON L AND P CO
163	SOUTH TEXAS PLANT-2	HOUSTON L AND P CO
164	BROWNS FERRY-1	T V A
165	BROWNS FERRY-2	T V A
166	BROWNS FERRY-3	T V A
167	SEQUOYAH-1	T V A
168	SEQUOYAH-2	T V A
169	WATTS BAR-1	T V A
170	WATTS BAR-2	T V A
171	BELLEFONTE-1	T V A
172	BELLEFONTE-2	T V A
173	HARTSVILLE-A1	T V A
174	HARTSVILLE-A2	T V A
175	HARTSVILLE-B1	T V A
176	HARTSVILLE-B2	T V A
177	PHIPPS BEND-1	T V A
178	PHIPPS BEND-2	T V A
179	YELLOW CREEK-1	T V A
180	YELLOW CREEK-2	T V A
181	COMANCHE PEAK-1	TEXAS UTIL GEN CO
182	COMANCHE PEAK-2	TEXAS UTIL GEN CO
183	DAVIS-BESSE-1	TOLEDO EDISON CO
184	DAVIS-BESSE-2	TOLEDO EDISON CO
185	DAVIS-BESSE-3	TOLEDO EDISON CO
186	CALLAWAY-1	UNION ELECTRIC CO
187	CALLAWAY-2	UNION ELECTRIC CO
188	VERMONT YANKEE	VERMONT YANKEE NUCL
189	SURRY-1	VIRGINIA E AND P CO
190	SURRY-2	VIRGINIA E AND P CO
191	NORTH ANNA-1	VIRGINIA E AND P CO
192	NORTH ANNA-2	VIRGINIA E AND P CO
193	NORTH ANNA-3	VIRGINIA E AND P CO
194	NORTH ANNA-4	VIRGINIA E AND P CO
195	WNP-2	WASH PUB PWR SUP SYS
196	WNP-1	WASH PUB PWR SUP SYS
197	WNP-3	WASH PUB PWR SUP SYS
198	WNP-4	WASH PUB PWR SUP SYS
199	WNP-5	WASH PUB PWR SUP SYS
200	POINT BEACH-1	WI EL PWR/WI MI PWR
201	POINT BEACH-2	WI EL PWR/WI MI PWR
204	KEWAUNEE	WISCONSIN P S CORP
205	YANKEE-ROWE	YANKEE ATOM ELEC CO
207	BRUNSWICK-2 PWR POOL	CAROLINA P AND L CO
208	BRUNSWICK-1 PWR POOL	CAROLINA P AND L CO
209	MILLSTONE-3 BWR POOL	NORTHEAST UTILITIES

APPENDIX F - BEST-CASE DATA

Shortly before the final printing of this work, all sources of input were reviewed and updated. The simulation model is exercised using these data. The results of this simulation are given in Chapter Five. This appendix contains the various input data along with the complete series of graphical output.

For the best-case simulation, a selective shipment priority between reactor and AFR facility is employed. The priority is based on the National Electric Reliability Council (NERC) codes. AFR Facility One (AGNS) receives fuel from all reactors within the Southeastern Electric Reliability Council (SERC).. AFR Facility Two (GE-Morris) accepts fuel from reactors within the East Central Area Reliability Coordination Agreement (ECAR), the Electric Reliability Council of Texas (ERCOT), the Mid-America Interpool Network (MAIN), the Mid-Continent Area Reliability Coordination Agreement (MARCA), the Southwest Power Pool (SWPP), and the Western Systems Coordinating Council (WSCC). Reactors within the Mid-Atlantic Area Council (MAAC) and the Northeast Power Coordinating Council (NPCC) ship fuel to AFR Facility Three, West

Valley. When any one of the above facilities becomes unavailable, the discharge is transferred to the nearest AFR site. Nearness is the shipping priority to both new AFR facilities and all repositories.

The accident analysis module is not utilized. The options to store fuel against FCR and track fuel within the temporary file are permitted. The repository scheduling is given in Table 3; the earliest scenario (i.e., first repository on-line in 1997) is employed. The domestic discharge data and the foreign data are listed in Tables 4 and 5, respectively.

The best-case data include two new AFR facilities. Table 18 describes the characteristics of each AFR facility and each repository. The AFR site data are accumulated through the tracking of the milestones within the AFR MIS. The final input block originates from the DISFUL program. Table 19 gives the domestic discharge figures.

Figures 37 through 48 are the utilization plots for the AFR facilities and repositories. An interpretation of these figures is given in Chapter Six.

Table 18. Storage Site Input Data for Best-case Simulation

STORAGE SITE	ON-LINE TIME	CAPACITY (mtu)	HANDLING RATE (mtu/year)	AMOUNT OF EXISTING SPENT FUEL	LATITUDE	LONGITUDE
BARNWELL	84.5	1750	750	0	33.2	81.4
GE-MORRIS	84.5	1100	250	350	41.4	88.4
GE-MORRIS Add-On	89.2	1700	250	0	41.4	88.4
WEST VALLEY	84.5	1700	750	164	42.4	78.7
NEW AFR 1	90.8	5000	1000	0	37.5	91.7
NEW AFR 2	93.8	5000	1000	0	35.0	110.0
REPOSITORY 1	97.6	41000	1800*	0	47.0	118.0
REPOSITORY 2	100.6	69000	1800*	0	33.0	110.0
REPOSITORY 3	103.6	69000	1800*	0	32.0	87.0
REPOSITORY 4	106.6	69000	1800*	0	43.0	74.0
REPOSITORY 5	109.6	69000	1800*	0	45.0	90.0

*After the first five years, this value increases to 6000

Table 19. Domestic Discharge Data for Best-case Simulation

REACTOR	ANNUAL MTU SHIPPED 1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999
1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	9	32
6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
7	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
8	0	0	0	0	0	0	18	21	21	21	21	21	21	21	21	21	21	21	21	21	21
9	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5	23	23	23	23	23	23
10	0	0	0	0	0	0	0	0	0	0	23	44	44	44	44	44	44	44	44	44	44
11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
12	0	0	0	0	0	0	0	0	0	0	0	18	22	22	22	22	22	22	22	22	22
13	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
14	0	14	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20	20
15	0	0	0	9	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27
16	0	0	26	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27
17	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	7
18	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
19	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
20	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
22	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	17	27	27	27	27	27
23	0	0	0	0	0	0	0	0	0	0	0	0	0	0	76	79	79	79	79	79	79
24	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
25	4	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
26	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	27	28	28	28
27	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	7	28	28	28
28	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	56	56	56	56
29	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
30	0	0	0	0	0	0	0	0	0	0	0	0	0	23	54	54	54	54	54	54	54
31	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
32	0	0	0	0	0	0	0	0	0	0	0	21	76	76	76	76	76	76	76	76	76
33	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
34	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	36	57	57	57	57	57
35	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
36	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	36	57	57	57	57	57
37	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
38	0	0	0	0	0	0	0	0	0	0	0	0	5	23	23	23	23	23	23	23	23
40	0	0	0	0	0	0	0	0	0	0	1	23	23	23	23	23	23	23	23	23	23
41	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3
42	0	0	0	0	4	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27	27
43	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	34	40	40	40	40	40
44	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
45	0	0	0	0	0	0	0	0	0	0	2	3	3	3	3	3	3	3	3	3	3
46	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	26	37	37	37	37	37
47	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
48	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
49	0	0	0	0	0	0	0	39	48	48	48	48	48	48	48	48	48	48	48	48	48
50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
51	77	24	0	0	0	0	0	17	24	24	24	24	24	24	24	24	24	24	24	24	24
52	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	14	28	28
53	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
54	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
55	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
56	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
57	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
58	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
59	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	11

Table 19. (continued)

REACTOR	ANNUAL MTU SHIPPED 1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999
60	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
61	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
63	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22	22	22	22	22	22	22
64	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	20
65	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	24	25	25
66	0	0	0	0	0	0	0	0	0	0	0	8	19	19	19	19	19	19	19	19	19
67	0	0	2	19	1	19	19	19	19	19	19	19	19	19	19	19	19	19	19	19	19
68	0	0	0	0	0	0	0	0	19	21	21	21	21	21	21	21	21	21	21	21	21
69	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	11	21	21	21	21
70	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	44	54	54
71	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
72	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
73	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
74	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
75	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
78	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	10	40	40
79	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4	32	32
80	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
81	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	53	55	55	55	55
82	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
83	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	18	18	18	18	18
85	0	0	0	0	0	8	21	21	21	21	21	21	21	21	21	21	21	21	21	21	21
86	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
87	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
88	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
89	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
90	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
93	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5	28	28	28	28
94	0	0	0	0	0	0	0	0	0	13	22	22	22	22	22	22	22	22	22	22	22
95	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6	21
96	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	43
98	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
99	0	0	0	0	0	0	0	0	0	0	0	26	26	26	26	26	26	26	26	26	26
100	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6	28	28
101	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6
102	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	10	31
103	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
104	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	12	21	21	21	21
105	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5	37	37
106	0	0	0	0	0	0	0	11	22	22	22	22	22	22	22	22	22	22	22	22	22
107	0	0	0	0	0	0	7	21	21	21	21	21	21	21	21	21	21	21	21	21	21
108	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	13	28	28	28	28
109	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
110	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
111	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
112	0	0	0	0	0	0	0	0	0	0	0	14	18	18	18	18	18	18	18	18	18
113	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5	27	27	27	27	27	27
114	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
116	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
117	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
118	0	0	0	0	0	0	0	0	0	0	5	13	13	13	13	13	13	13	13	13	13
119	0	0	0	0	0	0	0	0	0	2	3	3	3	3	3	3	3	3	3	3	3
120	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
121	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Table 19. (continued)

REACTOR	ANNUAL MTU SHIPPED 1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999
122	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	52	73	73	73	73
123	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
124	0	0	0	0	0	0	0	1	38	38	38	38	38	38	38	38	38	38	38	38	38
125	0	0	0	0	0	0	0	20	38	38	38	38	38	38	38	38	38	38	38	38	38
126	0	0	0	0	0	0	0	0	0	0	0	10	38	38	38	38	38	38	38	38	38
127	0	0	0	0	0	0	0	0	0	0	0	0	0	10	38	38	38	38	38	38	38
128	0	0	0	0	0	0	0	0	0	0	0	0	0	19	28	28	28	28	28	28	28
129	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
130	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
133	0	0	0	0	0	0	0	0	0	0	9	27	27	27	27	27	27	27	27	27	27
134	0	0	0	0	0	0	0	0	0	0	1	23	23	23	23	23	23	23	23	23	23
137	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6	56	56	56
138	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
139	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22	28	28
140	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22
141	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3	38	38	38	38	38
142	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	3	38	38	38	38
143	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	11	28	28	28	28
144	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	28	28	28	28
145	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
146	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
151	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	35	42	42	42
152	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	35	42
153	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	8	13	13	13	13	13
154	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
155	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	11	23	23	23	23	23
158	0	0	0	0	0	0	0	0	0	0	0	0	0	0	12	23	23	23	23	23	23
159	0	0	3	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11
160		0	0	0	0	0	0	0	0	0	0	0	12	28	28	28	28	28	28	28	28
161	0	0	0	0	0	0	0	0	0	0	0	0	0	12	28	28	28	28	28	28	28
162	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
163	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
164	0	0	0	0	0	0	0	0	0	0	0	0	78	81	81	81	81	81	81	81	81
165	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
166	0	0	0	0	0	0	0	0	0	0	0	6	40	40	40	40	40	40	40	40	40
167	0	0	0	0	0	0	0	0	0	0	0	0	47	56	56	56	56	56	56	56	56
168	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
169	0	0	0	0	0	0	0	0	0	0	0	0	18	56	56	56	56	56	56	56	56
170	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
171	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	30	30	30	30
172	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	30	30	30
173	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22	40	40	40	40
174	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22	40	40	40
175	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22	40	40	40	40
176	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22	40	40	40
177	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22	40	40	40
178	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22	40
179	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
180	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
181	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	33	57	57
182	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Table 19. (continued)

REACTOR	ANNAUL MTU SHIPPED 1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999
183	0	0	0	0	0	0	0	0	0	0	0	0	9	23	23	23	23	23	23	23	23
184	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
185	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
186	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
187	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
188	0	0	0	0	0	0	0	0	0	0	0	2	19	19	19	19	19	19	19	19	19
189	0	0	0	0	0	5	40	40	40	40	40	40	40	40	40	40	40	40	40	40	40
190	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
191	0	0	0	0	0	0	0	0	19	46	46	46	46	46	46	46	46	46	46	46	46
192	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
193	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
194	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
195	0	0	0	0	0	0	0	0	0	0	0	0	0	25	37	37	37	37	37	37	37
196	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	11	31	31
197	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	30	32
198	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	11	31
199	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	30
200	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	16	25
201	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
204	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
205	0	0	0	0	0	0	0	0	0	0	0	0	3	9	9	9	9	9	9	9	9
207	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
208	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
209	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

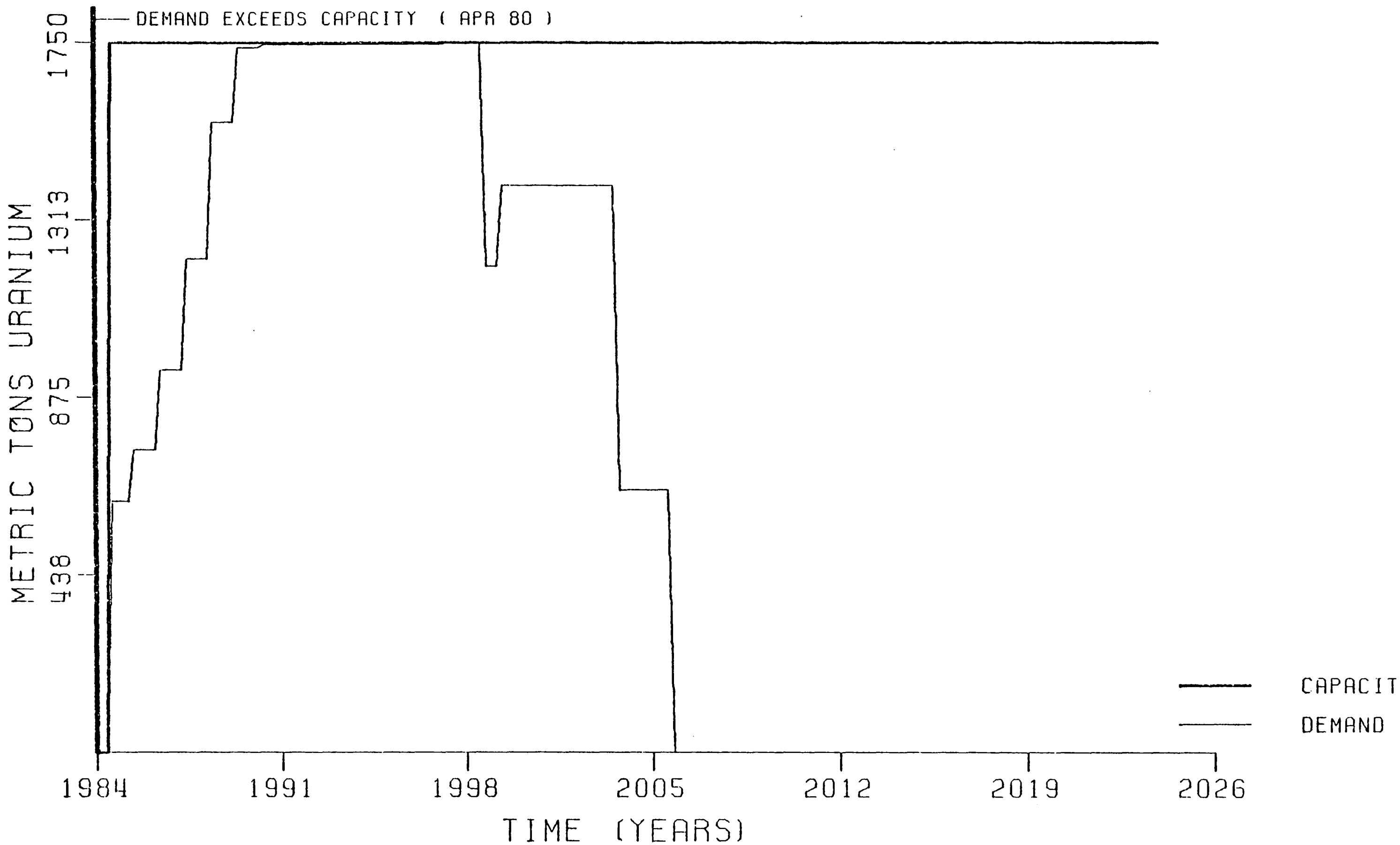

Figure 37. Storage Utilization for the Barnwell Facility
Shipment Mode to AFR: S
Based on Best-case Input Data

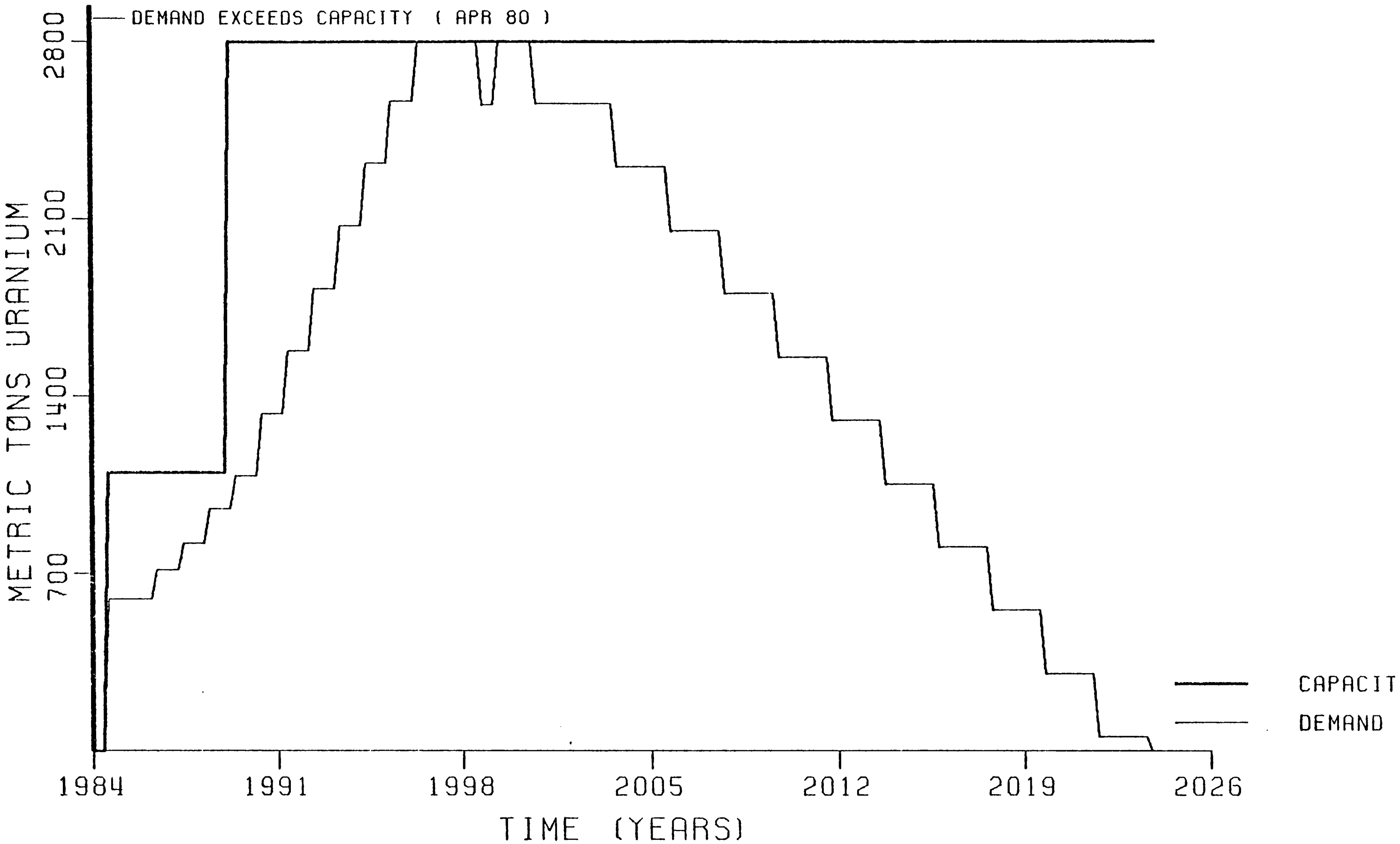

Figure 38. Storage Utilization for the Morris Facility
Shipment Mode to AFR: S
Based on Best-case Input Data

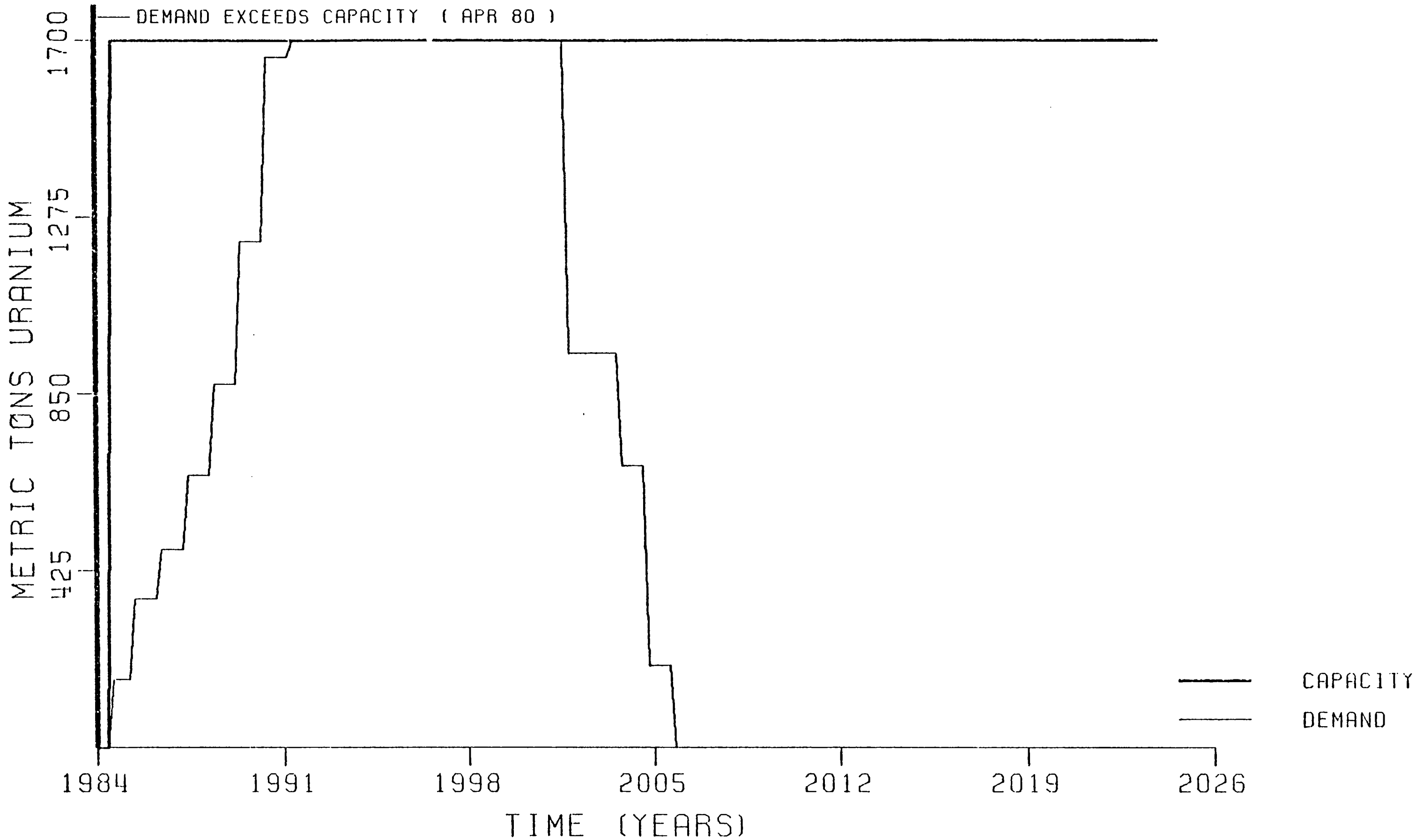

Figure 39. Storage Utilization for the West Valley Facility
Shipment Mode to AFR: S
Based on Best-Case Input Data

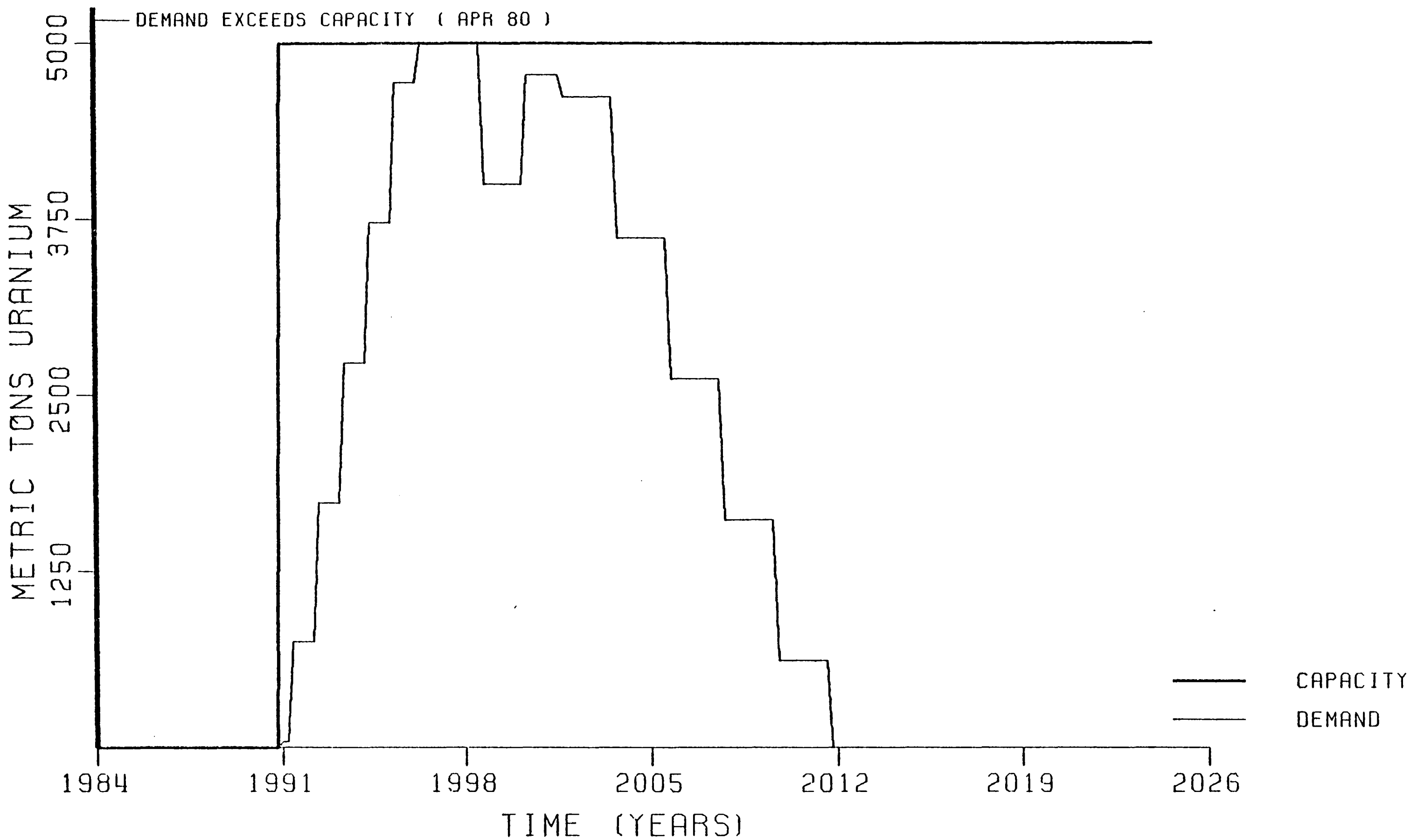

Figure 40. Storage Utilization for New AFR Facility 1
Shipment Mode to AFR: S
Based on Best-case Input Data

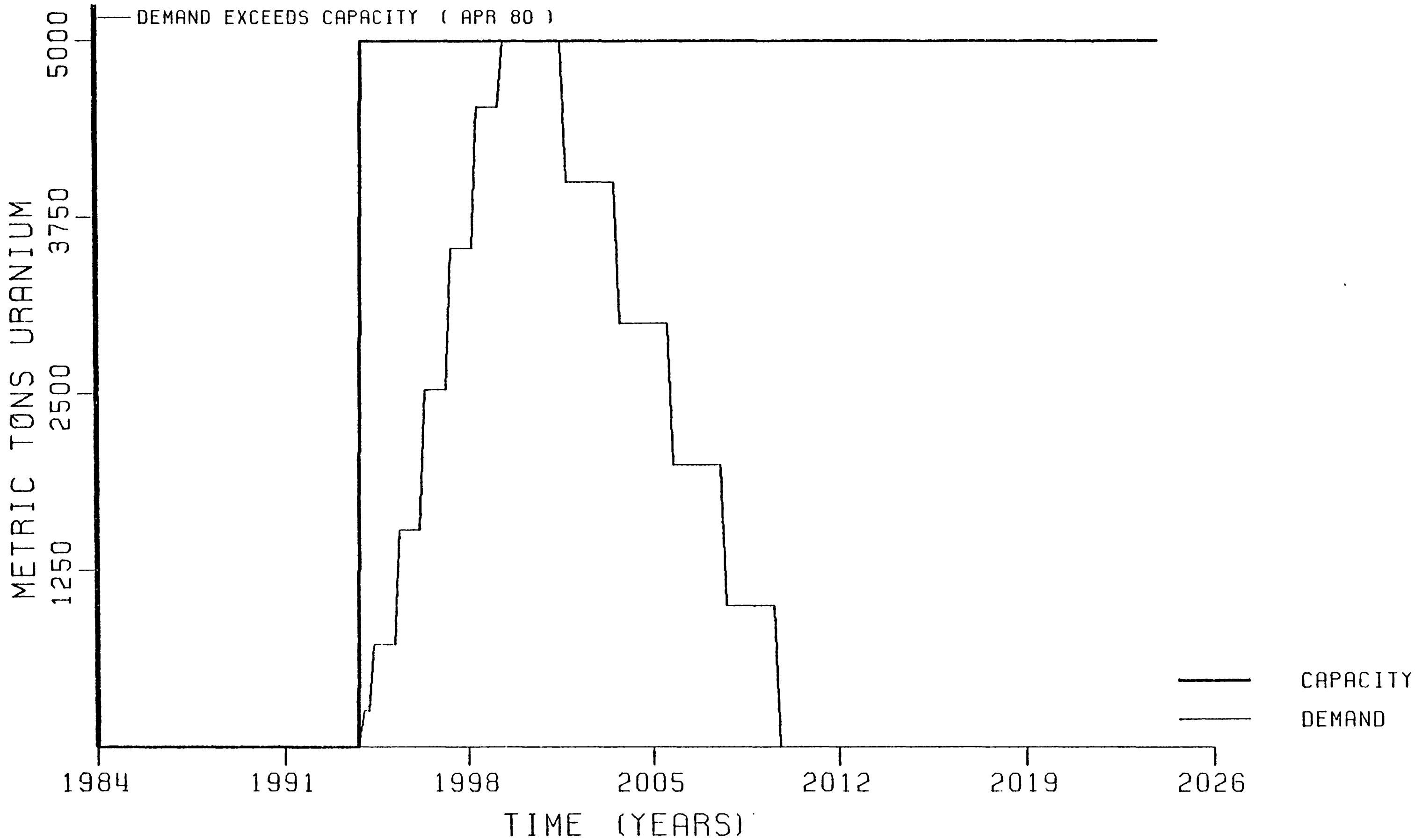

Figure 41. Storage Utilization for New AFR Facility 2
Shipment Mode to AFR: S
Based on Best-case Input Data

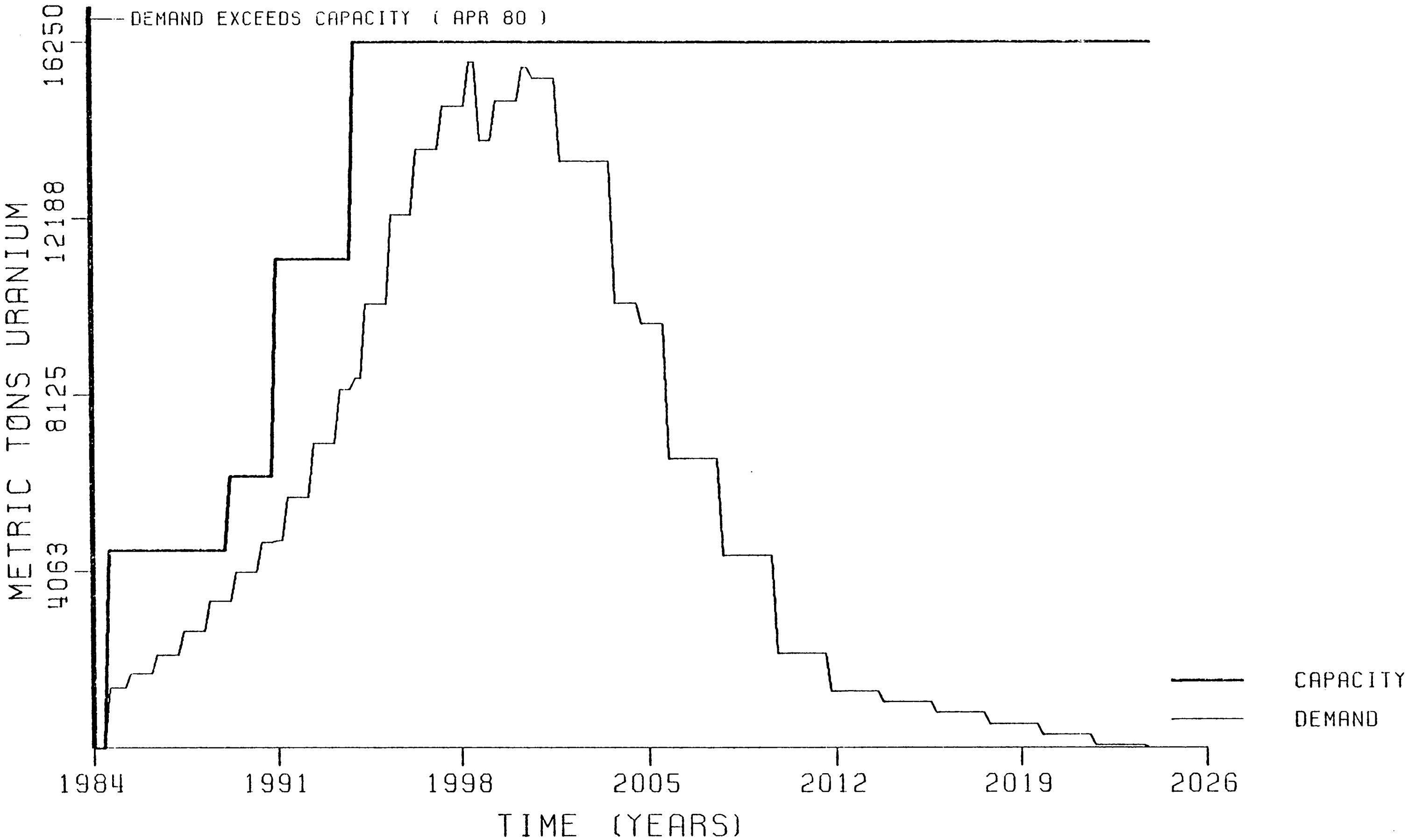

Figure 42. Storage Utilization for all AFR Facilities
Shipment Mode to AFR: S
Based on best-case Input Data

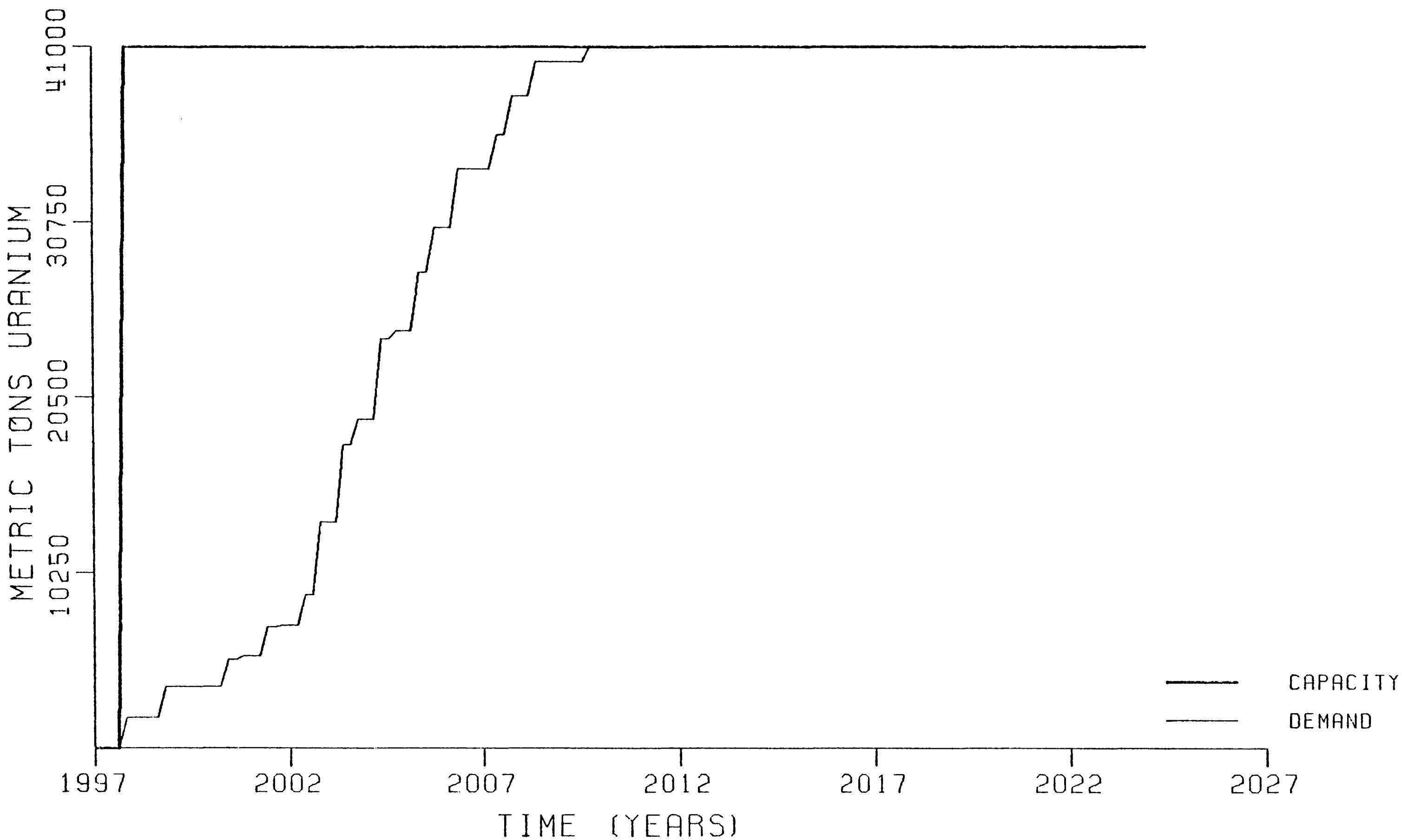

Figure 43. Storage Utilization for Repository Number 1
Shipment Mode to Repository: R
Based on Best-case Input Data

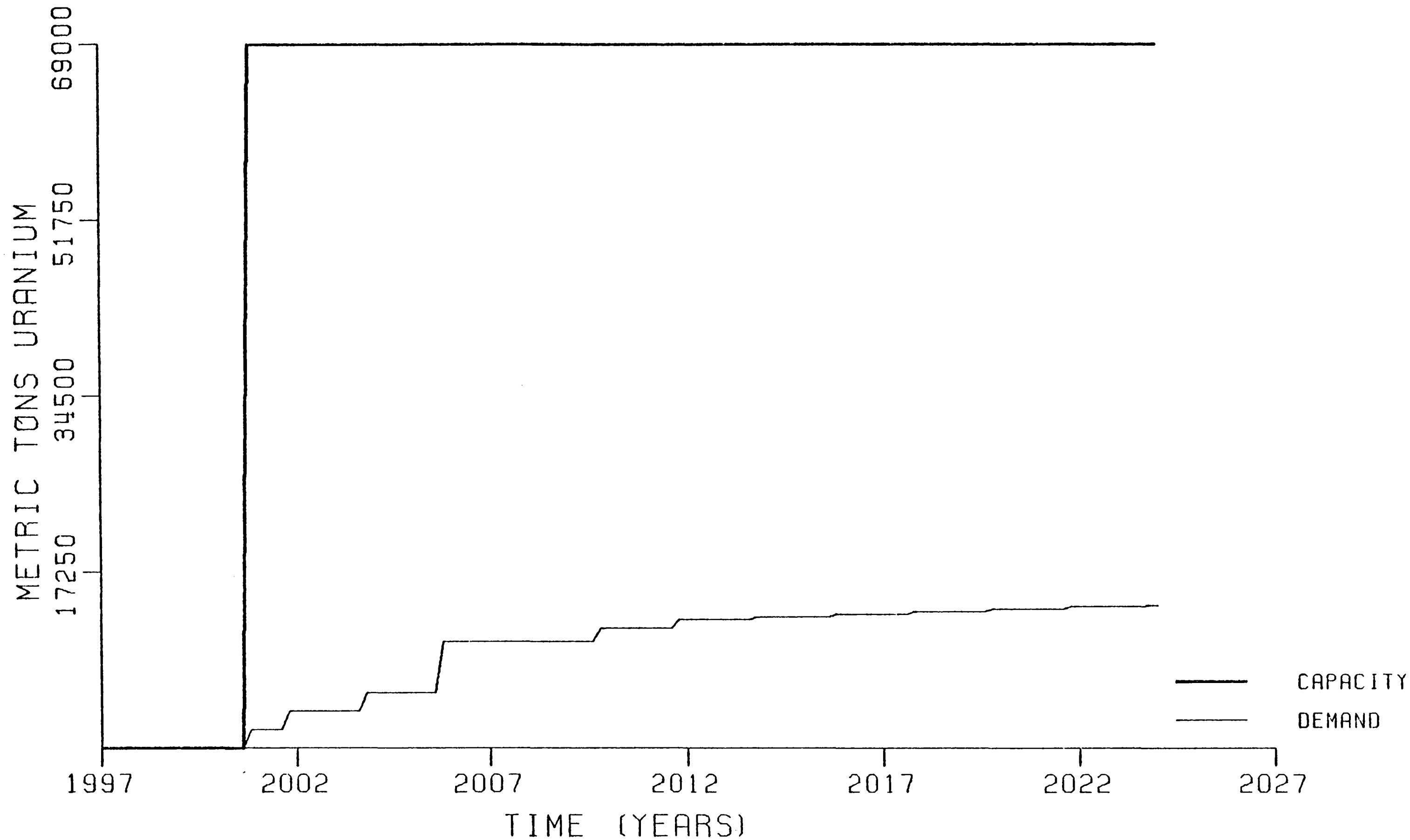

Figure 44. Storage Utilization for Repository Number 2
Shipment Mode to Repository: N
Based on best-case Input Data

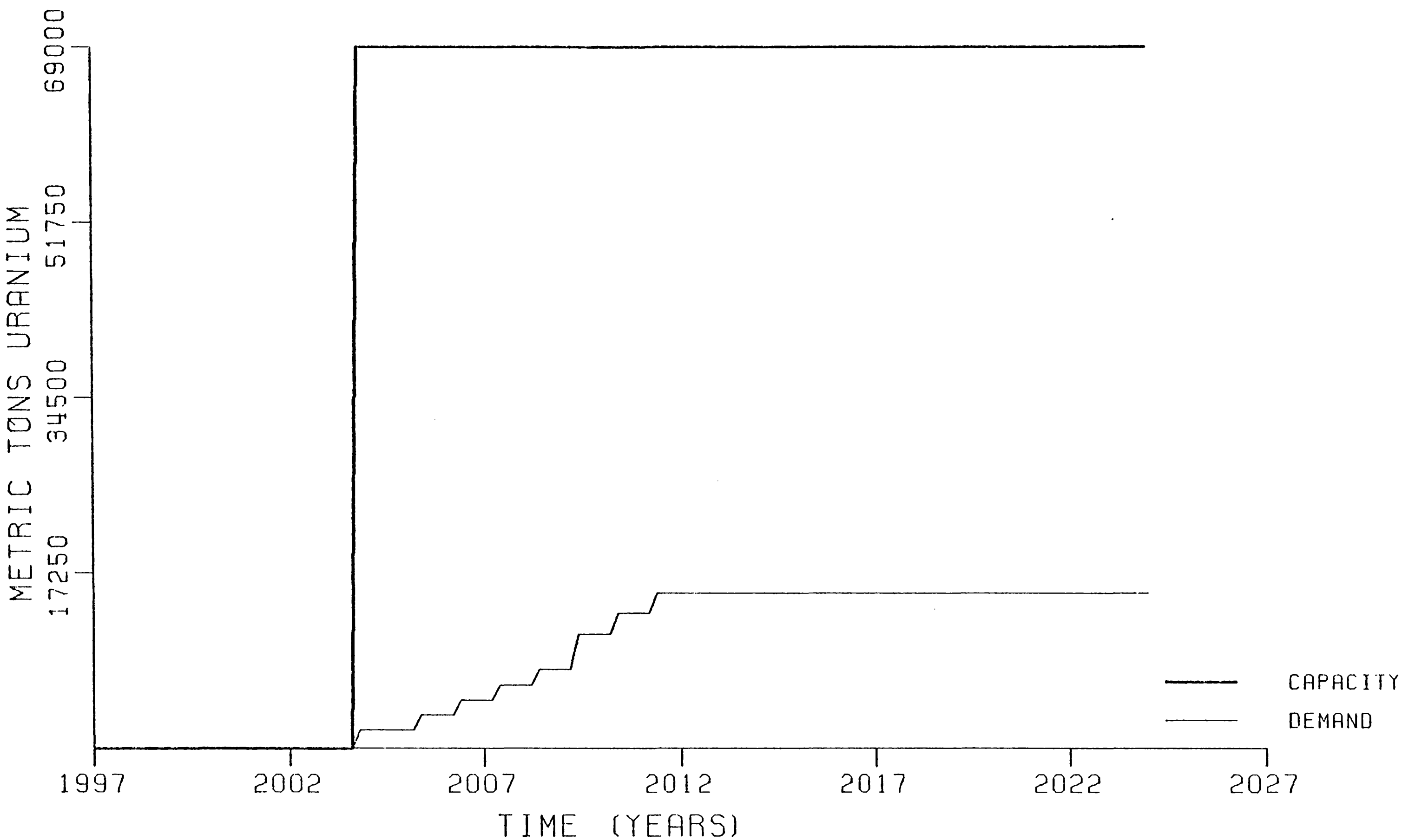

Figure 45. Storage Utilization for Repository Number 3
Shipment Mode to Repository: N
Based on Best-case Input Data

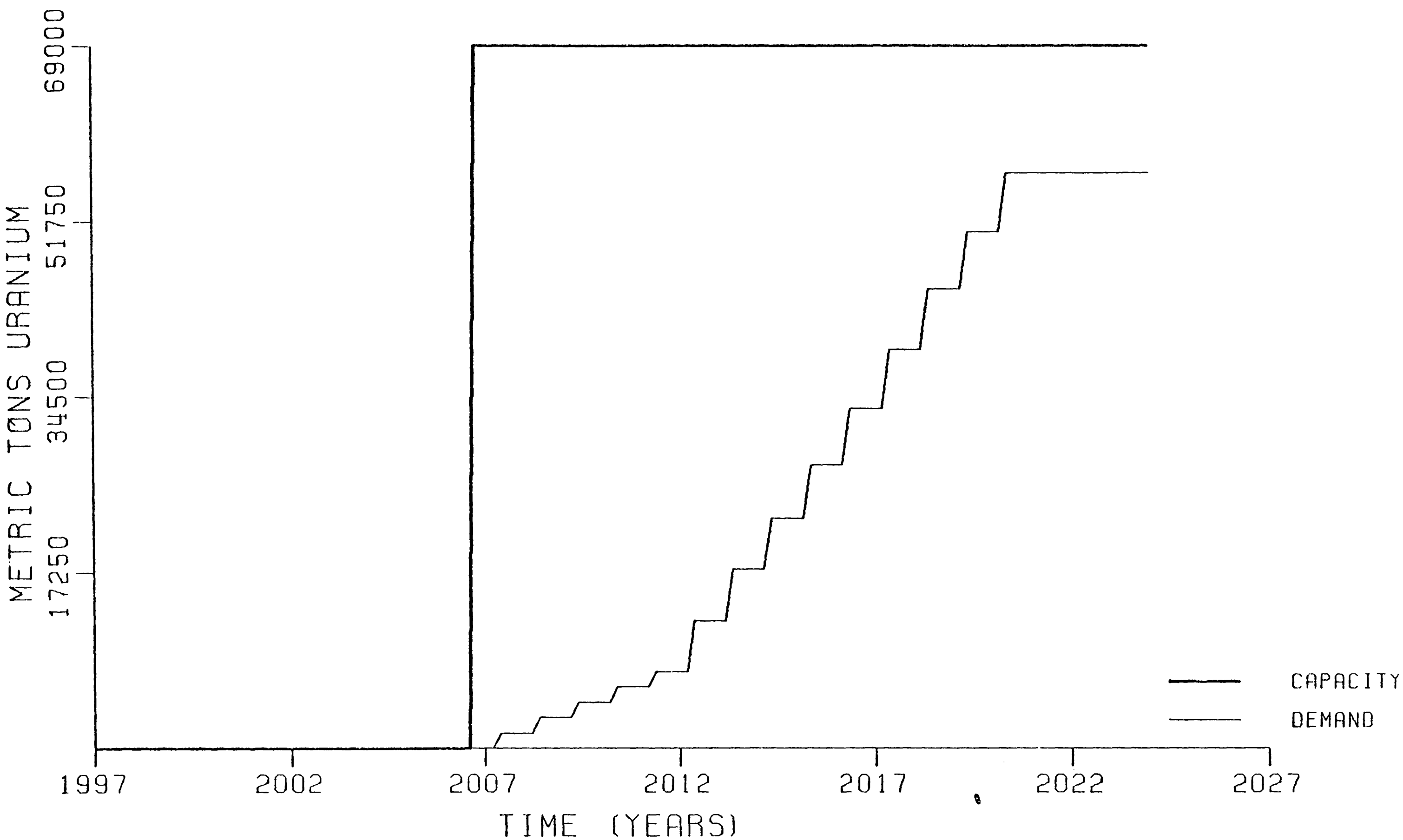

Figure 46. Storage Utilization for Repository Number 4
Shipment Mode to Repository: N
Based on Best-case Input Data

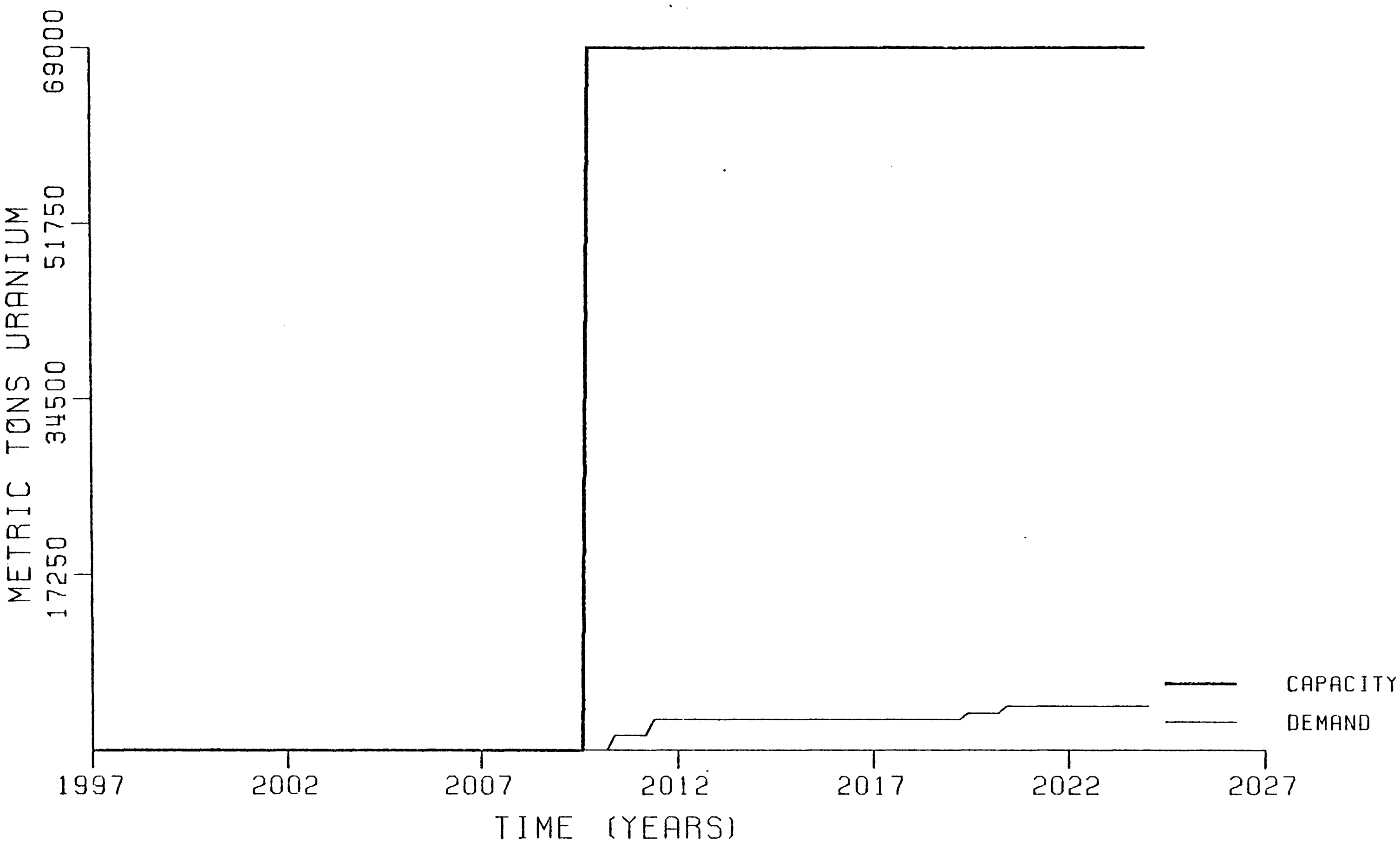

Figure 47. Storage Utilization for Repository Number 5
Shipment Mode to Repository: N
Based on Best-Case Input Data

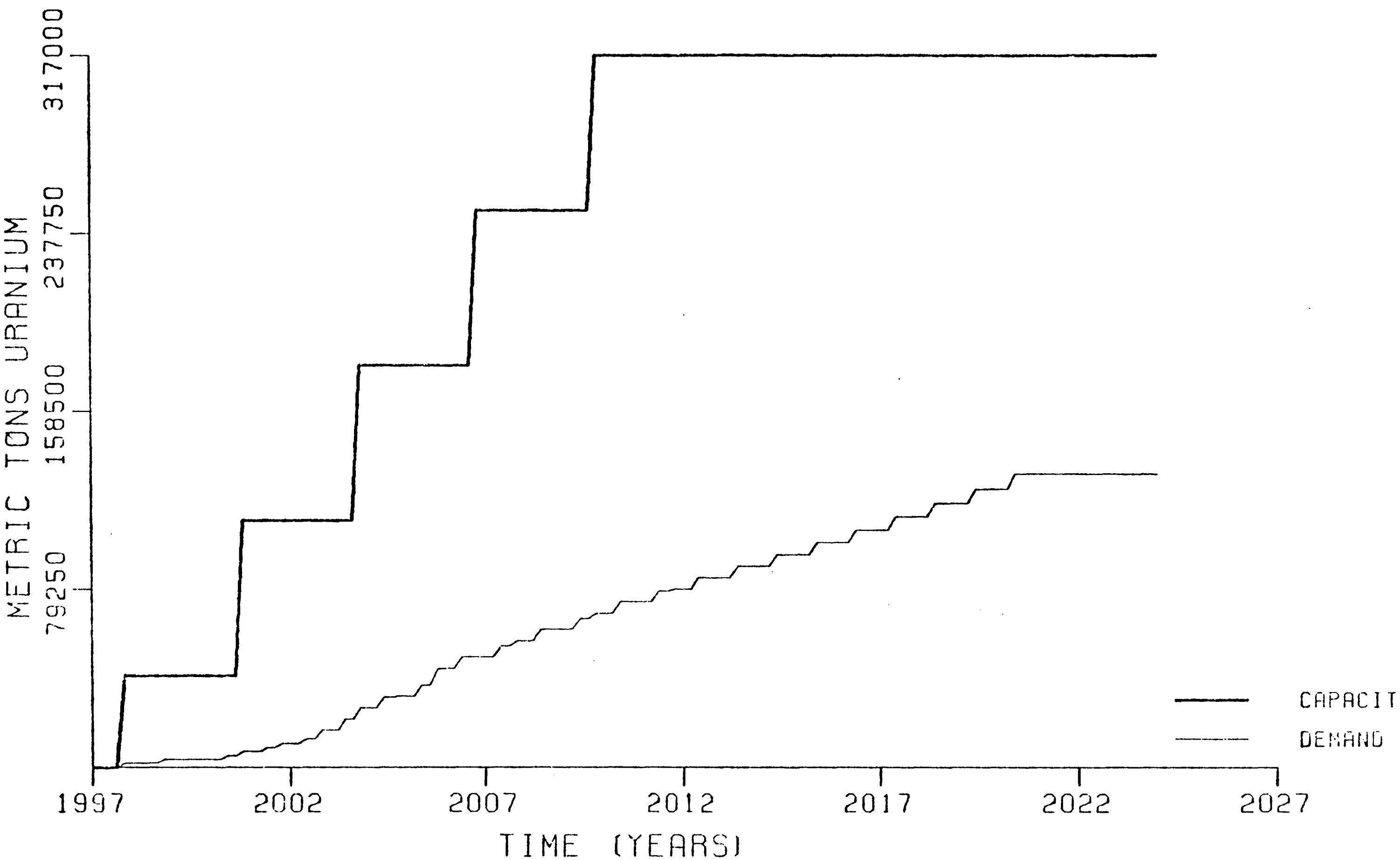

Figure 48. Storage Utilization for all Repositories
Shipment Mode to Repository: M
Based on Best-case Input Data

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ANALYSIS BY SIMULATION

OF THE DISPOSITION OF NUCLEAR FUEL WASTE

by

Jeffery Lee Turek

(ABSTRACT)

To achieve the non-proliferation objectives of the United States, the reprocessing of spent nuclear fuel was discontinued in 1977. Since current at-reactor storage capacity is based upon a nuclear fuel cycle which includes reprocessing, this halt in reprocessing is causing large quantities of non-storable spent fuel. Permanent nuclear waste storage repositories will not be available until the end of the century. Present Department of Energy policy calls for sufficient interim Away-From-Reactor (AFR) Storage capacity to insure that no commercial reactor has to shutdown due to inadequate storage space for discharged spent fuel.

A descriptive simulation model is developed which includes all aspects of nuclear waste disposition. The model is comprised of two systems, the second system orchestrated by GASP IV. A spent fuel generation prediction module is interfaced with the AFR Program

Management Information System and a repository scheduling information module. The user is permitted a wide range of options with which to tailor the simulation to any desired storage scenario. The model projects storage requirements through the year 2020.

The outputs are evaluations of the impact that alternative decision policies and milestone date changes have on the demand for, the availability of, and the utilization of spent fuel storage capacities. Both graphs and detailed listings are available. These outputs give a comprehensive view of the particular scenario under observation, including the tracking, by year, of each discharge from every reactor.

Included within the work is a review of the status of spent fuel disposition based on input data accurate as of August 1980. The results indicate that some temporary storage techniques (e.g., transshipment of fuel and/or additional at-reactor storage pools) must be utilized to prevent reactor shutdowns. These techniques will be required until the 1990's when several AFR facilities, and possibly one repository, can become operational.