INTERN EXPERIENCE AT

ARIZONA PUBLIC SERVICE COMPANY

AN INTERNSHIP REPORT

by

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ABSTRACT

Intern Experience at Arizona

Public Service Company (July 1986)

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This report is a description of the author's experience as an intern with the Arizona Nuclear Power Project. For the duration of the internship period, the author worked as an Engineer I in the Technical Projects Section of the Nuclear Fuel Management Department.

During the internship period, the author was assigned three major tasks. The first of these tasks was to develop a computer code to predict the number of failed fuel rods based upon the response of the let-down process radiation monitor. The second task was to identify and procure a computer code which best fulfilled the needs of the company for forecasting the requirements, costs and cash flows associated with the procurement of nuclear fuel. The third major task assigned to the author was researching the relevant issues and developing a basis from which to negotiate the cost responsibility with Combustion Engineering for obtaining additional thermal margin. In addition to these major tasks, the author was also given many less substantial assignments in a wide variety of areas for which the Technical Projects Section is responsible.

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INTRODUCTION

This report documents the experience accrued by the author during a twelve month Doctor of Engineering internship served with the Arizona Public Service Company, the project manager and operating agent for the Arizona Nuclear Power Project. The author served the internship between June 1, 1984 and May 31, 1985, as an Engineer I assigned to the Technical Projects Section. Included in this report are the objectives of the internship and details of a portion of the work performed by the author during the internship.

Internship Objectives

The overall objectives of the internship are as outlined in the Doctor of Engineering manual. These are:[1]

a. To enable the student to demonstrate and enhance his abilities to apply both knowledge and technical training by making an identifiable contribution in an area of practical concern to the organization in which the Internship is served.

b. To enable the student to function in a non-academic environment in a position in which he will become aware of the employer's approach to problems.

Utilizing these general guidelines and the advice of the internship supervisor, the author formulated a set of specific internship objectives. These were:[2]

A. Fuel Management Objectives

1. Learn to use and understand industry fuel management computer codes for core design and operations support

activities.

2. Learn to evaluate the need for, operation of and costs of using the SAROS computer code.

3. Learn the philosophy behind successful management of company resources and assets; specifically, management of nuclear fuel for the Palo Verde nuclear reactors.

4. Interact with the nuclear fuel vendors and various engineering service organizations to successfully accomplish fuel management activities.

B. Fuel Cost Predicting And Accounting Objectives

1. Learn to perform the necessary economic and technical analysis to support fuel cost forecasting and accounting.

2. Investigate existing software packages which perform these functions and recommend one for implementation.

3. Learn the requirements of and uses for the software package for each department of the company and each of the Palo Verde project participants.

4. Interact with the various departments of the company and of the participants' Engineering and Operations Committee to implement the selected software package.

C. Personal and Professional Objectives

1. Interact with the various groups within the company, contractors, the participants and other supporting organizations to increase the author's communication skills.

2. Participate in professional activities such as state and national engineering societies.

Internship Organization

Arizona Nuclear Power Project

The Arizona Nuclear Power Project (ANPP) was formed in April of 1972 to engineer, design, construct, license and operate the Palo Verde Nuclear Generating Station (PVNGS). Currently, the project is a joint effort of six utility companies who share construction and operating expenses as well as the electricity which is generated. Arizona Public Service Company (APS) is both project manager and operating agent for ANPP. Each of the participants in the project are listed in Table 1 as are their respective percentages of ownership.

Palo Verde Nuclear Generating Station

The Palo Verde Nuclear Generating Station is located on a 4,050 acre site approximately 55 miles west of Phoenix, near the small town of Wintersburg, Arizona. PVNGS is comprised of three nearly identical 1,275 megawatt pressurized water reactors and a Water Reclamation Facility for the treatment of sewage effluent which is ultimately used for condenser cooling water. Each of the nuclear steam supply systems was designed and constructed by Combustion Engineering, Inc. (CE). Bechtel Power Corporation served as the architect/engineer and construction manager for the project. When all three units are completed, PVNGS will be the largest nuclear power station in the United States.

Arizona Public Service Company

Arizona Public Service Company is one of the fully owned

TABLE 1

ANPP PARTICIPANTS AND OWNERSHIP SHARE

Participants	Percentage of Ownership
Arizona Public Service Company	29.1%
Salt River Project	23.19%*
Southern California Edison	15.8%
El Paso Electric Co.	15.8%
Public Service Co. of New Mexico	10.2%
Southern California Public Power Authority	5.91%

*At the time commercial operations begin, Salt River Project will transfer 5.7% of its interest in the project to the Los Angeles Department of Water and Power.

subsidiaries of the AZP Group, Inc. The structure of the AZP Group is depicted in Figure 1. APS is a utility which provides electrical service to over 500,000 customers. With an installed generating capacity of approximately 3,300 megawatts, APS serves over one-half the residents of the state. At present, the company is a fossil fuel based utility but as PVNGS begins commercial operation, a significant portion of the electricity generated will be from nuclear power.

Acting as operating agent for ANPP, APS has established a large organization dedicated exclusively to the support of ANPP and its participants. Figure 2 depicts the structure of this organization.

Nuclear Fuel Management Department

The Nuclear Fuel Management Department was formed during a company-wide reorganization shortly before the author's internship began. The department is comprised of four sections:

1. Nuclear Analysis
2. Safety Analysis
3. Fuel Cycle Services
4. Technical Projects

These four sections deal with matters relating to fuel performance monitoring and analysis, safety analysis, core physics analysis, reload planning and specification, fuel procurement, fuel fabrication, nuclear fuel cost forecasting and allocation, operational support analysis and all other fuel specific issues. The organization of the department as well as the position the author occupied are illustrated in Figure 3.

FIGURE 1

STRUCTURE OF THE AZP GROUP, INC.

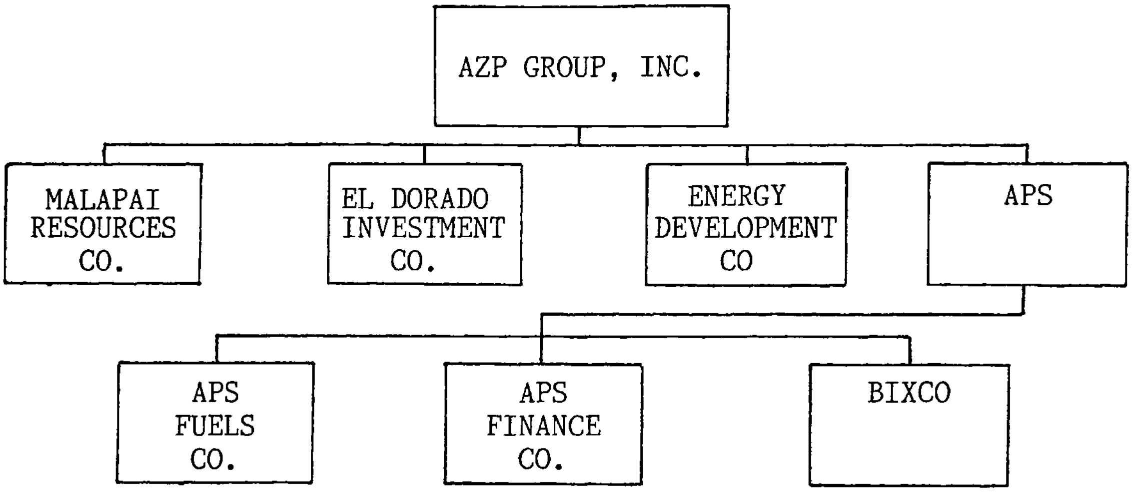


FIGURE 2

ARIZONA NUCLEAR POWER PROJECT ORGANIZATION

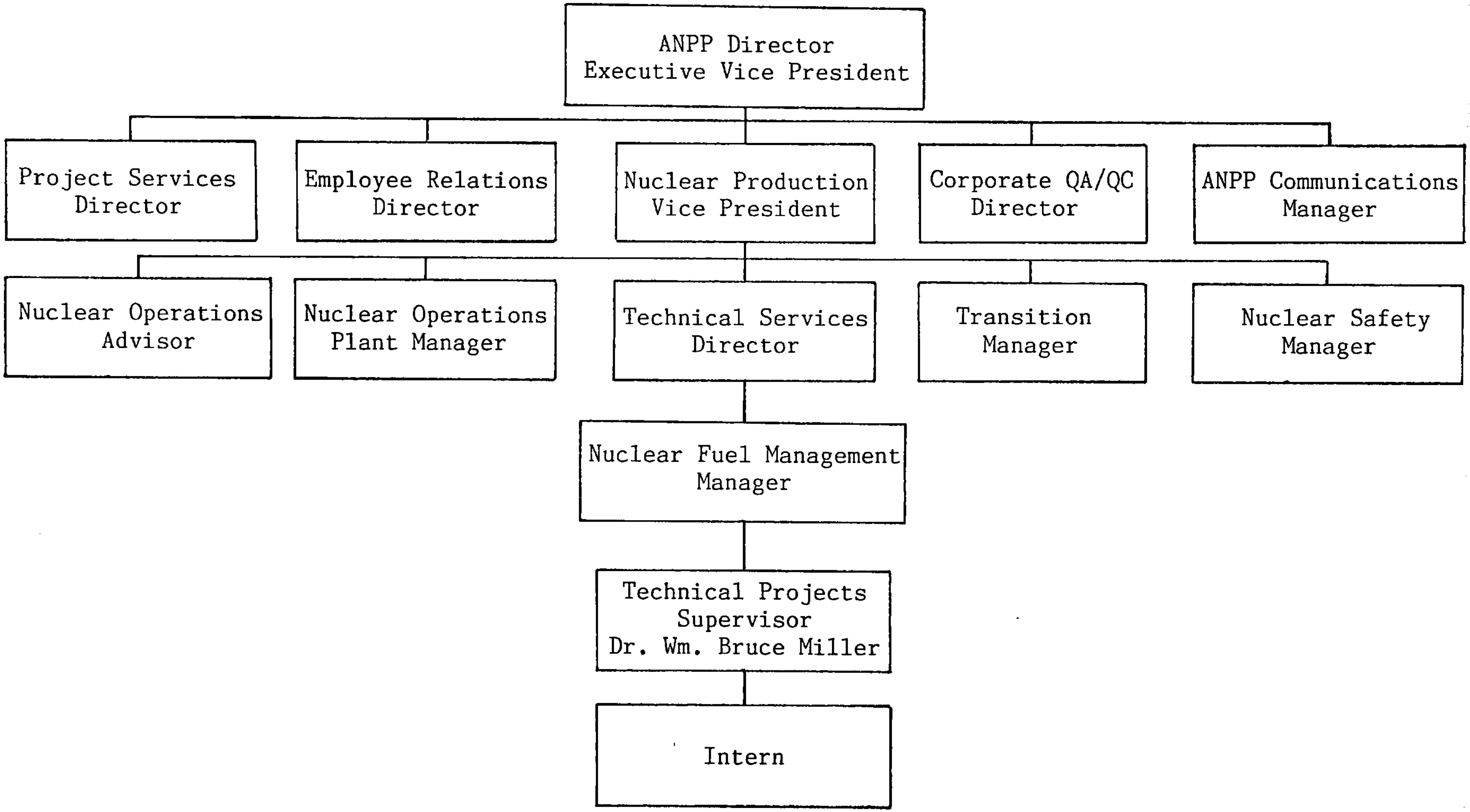
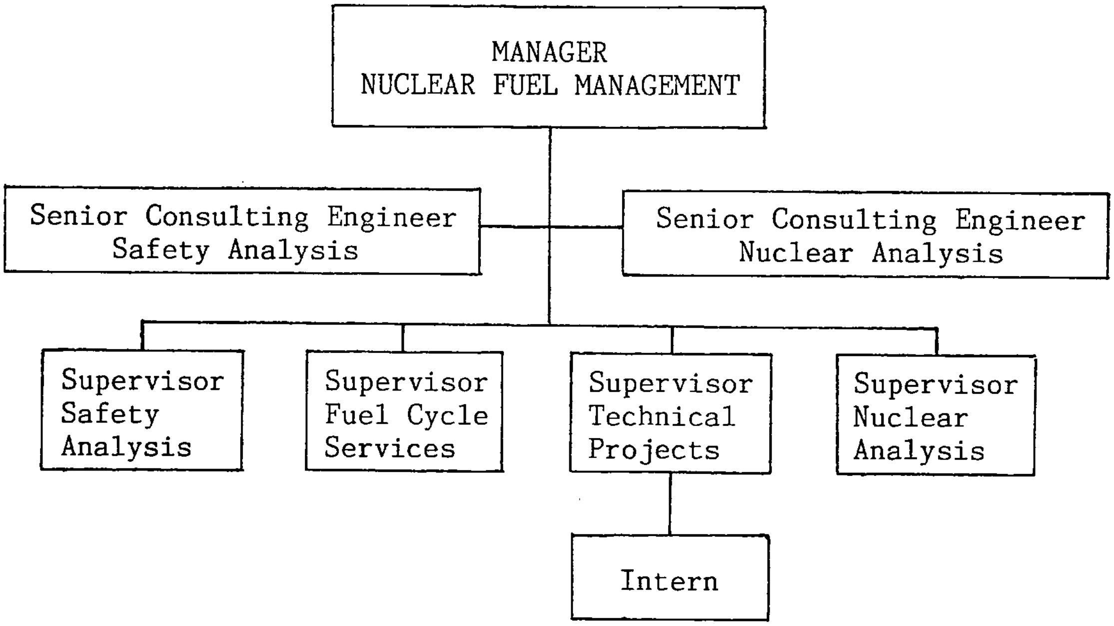


FIGURE 3

NUCLEAR FUEL MANAGEMENT DEPARTMENT ORGANIZATION



During the author's internship, the Nuclear Fuel Management staff grew rapidly. The department's staff was comprised primarily of nuclear engineers with a complement of technical analysts and clerical support personnel. The overall employment level for the department was projected to be 48 individuals by the end of 1985.

<u>Technical Projects Section</u>

The Technical Projects Section is supervised by Dr. William Bruce Miller who also served as the internship supervisor. The technical responsibilities of the group include:[6]

1. nuclear fuel fabrication
2. fuel vendor surveillance and performance evaluation
3. fuel-related operation recommendations and guidelines
4. fuel warranty and vendor supplied restrictions compliance
5. fuel surveillance and examination program development
6. fuel performance follow
7. core protection and monitoring system software specification, acquisition, evaluation, implementation and change control
8. reload planning and specification
9. reload design report review
10. Technical Specification, set point, and software update review
11. fuel vendor interface
12. fuel vendor transition program
13. reload data management program
14. fuel technology evaluation

Organizations that the group frequently interface with include:

1. Combustion Engineering
2. Westinghouse
3. U.S. Nuclear Regulatory Commission
4. ANPP Licensing Department
5. ANPP Quality Assurance Department
6. PVNGS Reactor Engineering Section
7. PVNGS Operations Department
8. ANPP Participant Services Department
9. ANPP Nuclear Engineering Department
10. Other elements of ANPP and APS
11. Other nuclear utilities

Working with the Technical Projects Section during the author's internship was both an interesting and challenging experience. The author was able to make direct contributions to the solution of several problems which were of practical concern to ANPP. Through observation and numerous interactions with various levels of management, the author gained an appreciation for ANPP's methodology for the resolution of problems and the general conduct of business.

FUEL FAILURE CORRELATION

When the author began his internship, ANPP was working vigorously to complete those remaining items required to obtain PVNGS Unit 1's operating license. One of these items, completion of the Emergency Plan, still required significant work at that time. One of the sub-tasks for finalizing the Emergency Plan was defining the criteria to be utilized to determine the classification of postulated abnormal events and accidents based upon the perceived threat to the health and safety of the public. In particular, one of the criteria which the Emergency Planning Department wanted to utilize was the failure of one or more percent of the fuel rods in the core. This criteria though was deemed to be less than desirous since the estimation of the fraction of failed fuel is an indirect process. To facilitate the easy implementation of this procedure by the plant operations staff, it was decided to correlate the let-down line process radiation monitor's response to the fraction of failed fuel in the core.

Nuclear Fuel Management was requested by the Emergency Planning Department to develop this correlation and the supporting methodology. Subsequently, the author was assigned as the Responsible Engineer for this task. A brief description of the PVNGS Emergency Plan, the failed fuel prediction model and the results from implementing the model follow.

PVNGS Emergency Plan

The overall objective of the Emergency Planning Department is to effectively protect the health and safety of the public during abnormal events which may occur at PVNGS. To accomplish this objective the Emergency Plan and the Emergency Plan Implementing Procedures have been developed. These documents provide the operations personnel at PVNGS with effective tools to mitigate the consequences of any emergency situation.

The overall plan is comprised of five major components. These are:

1. the appropriate classification of abnormal events.
2. the basis for classification of abnormal events.
3. the development of Emergency Plan Implementing Procedures.
4. the development of a system for maintaining effective Emergency Plan Implementing Procedures.
5. the development of interfaces with appropriate offsite agencies and authorities.

The Plan also has several key interfaces with other programs such as the Recovery Operations Program. A brief explanation of the portions of the Emergency Plan which are germain to the author's assignment follow.

The first step delineated in the Emergency Plan is to select the appropriate classification for the abnormal event. The four classifications contained in the Emergency Plan and a brief description of their meaning are:

1. Notification of Unusual Event - An event which indicates a potential degradation of the level of safety of the plant. No significant releases of radioactive material are expected to occur.

2. Alert - An event which involves an actual or potential substantial degradation of the level of safety of the plant. Any releases of radioactive material are expected to be small fractions of the Environmental Protection Agency Protective Action Guidelines.
3. Site Area Emergency - An event which involves actual or likely major failures of plant functions needed for the protection of the public. Any releases of radioactive material are not expected to exceed the Environmental Protection Agency Protective Action Guidelines except near the site boundary.
4. General Emergency - An event which involves actual or imminent substantial core degradation or melting concurrent with the potential for loss of containment integrity. Releases of radioactive material are expected to exceed the Environmental Protection Agency Protective Action Guidelines offsite.

The selection process for selecting the appropriate classification is primarily based on the status of the three main barriers to the release of radioactive material. These are fuel cladding integrity, primary coolant system boundary integrity and containment integrity. Table 2 correlates the classifications to the status of the three main barriers. Table 3 provides the criteria which are utilized in determining the status of the barriers.[7]

Since no direct method for determining the integrity of the fuel rod cladding during abnormal events exist, indirect methods such as

TABLE 2

CORRELATION OF ABNORMAL EVENT CLASSIFICATION

TO THE STATUS OF THE FISSION PRODUCT BARRIERS

Classification	Status of Barriers
Notification of Unusual Event	All three barriers are intact.
Alert	Two barriers are intact, one barrier has been verified as failed.
Site Area Emergency	One barrier intact, two barriers have been verified as failed.
General Emergency	All three barriers have been verified as failed.

TABLE 3

FAILURE CRITERIA FOR THE THREE MAIN

FISSION PRODUCT BARRIERS

Fission Product Barrier	Failure Criteria
Fuel Cladding	Greater than one percent of the fuel rods have perforated cladding.
Primary Coolant System Boundary	Greater than a 50 gallon per minute leak of primary coolant.
Containment	Greater than a 0.10 percent by weight leak of containment air per 24 hours at any pressure up to the design limit of 49.2 psig.

radio-chemical analysis or correlating to the primary coolant system specific activity must be employed. The Emergency Planning Department wanted to utilize the let-down process radiation monitor to infer the primary coolant system activity and thereby predict the number of failed fuel rods. Development of this methodology was ultimately assigned to the author.

Analytical Model Development

After investigating the current state of the art for correlating the number of failed fuel pins to primary system coolant activities, the author elected to develop a simplified model to perform preliminary scoping studies. If the results of this study were positive, a more detailed model would then be developed for actual use. The basic model assumed that the total primary system activity following a severe transient is due to four sources. They are:

1. the expected primary system activity during normal plant operations.
2. the expected "spiking" of activity caused by the thermal transient.
3. the release of the failed fuel rod gap's fission product inventory.
4. the release of the fuel pellet fission product inventory through a diffusion process.

The activities associated with each of these four components is comprised of many individual isotopes. The decay of each of these isotopes is explicitly considered in the model.

After the basic phenomena to be modelled were established, a number of simplifying assumptions were made. These are:

1. Complete and instantaneous mixing of the fission products with the primary system coolant.
2. During the transient, let-down to the Chemical and Volume Control System would be isolated. Let-down and, therefore, clean-up of the primary system coolant might be re-established at a later time.
3. The fuel failure mechanisms would be limited to clad rupture due to internal over pressurization. No fuel pellet over-heating or pellet melting was considered.
4. The fission product inventories were assumed to be end-of-cycle values in an equilibrium core. If the results of the scoping study were favorable, a method of adjusting the inventories to reflect the actual power history of the core would be incorporated in the more detailed final model.
5. A total release of the failed fuel rod's gap inventory was assumed.
6. Release of the failed fuel pellet's inventory was modeled by an escape rate coefficient method.
7. No plate-out or other losses of fission products from the primary system coolant were considered.
8. No dilution of the specific activity of the primary coolant was assumed (i.e. no actuation of the High or Low Pressure Safety Injection System was assumed).
9. Only the isotopes listed in Table 4 were considered in the scoping calculations.

TABLE 4

LIST OF ISOTOPES CONSIDERED IN THE SCOPING STUDY

Isotope	Half Life
I-131	8.041 days
I-132	2.285 hours
I-133	20.8 hours
I-134	52.6 minutes
I-135	6.585 hours
Kr-85M	4.48 hours
Kr-85	10.73 years
Kr-87	76.0 minutes
Kr-88	2.80 hours
Xe-131M	11.99 days
Xe-133	5.29 days
Xe-135	9.17 hours
Xe-138	14.2 minutes

Utilizing these simplifying assumptions, the basic scoping model was constructed. The set of differential equations and associated boundary conditions (for one isotope) that comprise the model are:

$$\frac{d}{dt}\left[A_c(t)\right] = \begin{cases} -\lambda A_c(t) & o\leq t\leq t_o \\ -(\lambda+k_p)A_c(t) & t\geq t_o \end{cases} \quad (1)$$

$$\frac{d}{dt}\left[A_s(t)\right] = \begin{cases} -\lambda A_s(t) & o\leq t\leq t_o \\ -(\lambda+k_p)A_s(t) & t\geq t_o \end{cases} \quad (2)$$

$$\frac{d}{dt}\left[A_g(t)\right] = \begin{cases} -\lambda A_g(t) & o\leq t\leq t_o \\ -(\lambda+k_p)A_g(t) & t\geq t_o \end{cases} \quad (3)$$

$$\frac{d}{dt}\left[A_p(t)\right] = -(\lambda+\nu)A_p(t) \qquad t\geq o \quad (4)$$

$$\frac{d}{dt}\left[A_f(t)\right] = \begin{cases} -\lambda A_f(t) + \nu A_p(t) & o\leq t\leq t_o \\ -(\lambda+k_p)A_f(t) + \nu A_p(t) & t\geq t_o \end{cases} \quad (5)$$

$$\frac{d}{dt}\left[A_t(t)\right] = \frac{d}{dt}\left[A_c(t)\right] + \frac{d}{dt}\left[A_s(t)\right] + \frac{d}{dt}\left[A_g(t)\right] + \frac{d}{dt}\left[A_f(t)\right] \qquad t\geq o \quad (6)$$

and

$$A_s(o) = R\cdot A_c(o) \quad (7)$$

$$A_p(o) = C\cdot Y\cdot A_{pp}(o) \quad (8)$$

$$A_g(o) = C\cdot Y\cdot A_{gg}(o) \quad (9)$$

$$A_f(o) = 0 \quad (10)$$

where:

A_c is the coolant specific activity due to fission products normally found in the primary system coolant,

A_s is the coolant specific activity due to spiking following a thermal transient,

A_g is the coolant specific activity due to the release of the failed fuel gap's fission product inventory,

A_p is the activity contained in the failed fuel pellets,

A_f is the coolant specific activity due to fission products diffusing out of the failed fuel pellets,

A_t is the total coolant specific activity,

λ is the decay constant,

k_p is the clean-up constant,

t is the elapsed time since the transient,

t_o is the elapsed time since the transient when let-down (clean-up) is re-established,

ν is the fuel pellet fission product escape rate coefficient which is defined as the fraction of the fuel pellet fission product inventory that diffuses out of the pellet per unit of time,

C is a constant which converts activity released into the primary coolant system to coolant specific activity,

R is the spiking ratio which is the ratio of the coolant specific activity following a thermal transient to the activity preceding the transient,

Y is the fraction of the total fuel rods which are assumed to have failed,

A_{gg} is the total activity contained in all the fuel rod's gaps.

A_{pp} is the total activity contained in the fuel pellets.

The solution to the set of differential equations is:

$$A_c(t) = \begin{cases} A_c(o)e^{-\lambda t} & o \leq t \leq t_o \\ A_c(t_o)e^{-(\lambda+k_p)(t-t_o)} & t \geq t_o \end{cases} \qquad (11)$$

$$A_s(t) = \begin{cases} R \cdot A_c(o)e^{-\lambda t} & o \leq t \leq t_o \\ A_s(t_o)e^{-(\lambda+k_p)(t-t_o)} & t \geq t_o \end{cases} \qquad (12)$$

$$A_g(t) = \begin{cases} C \cdot Y \cdot A_{gg}(o)e^{-\lambda t} & o \leq t \leq t_o \\ A_g(t_o)e^{-(\lambda+k_p)(t-t_o)} & t \geq t \end{cases} \qquad (13)$$

$$A_f(t) = \begin{cases} \dfrac{C \cdot Y \cdot \nu \cdot A_{pp}(o)}{\lambda+\nu}\left[1-e^{-\nu t}\right]e^{-\lambda t} & o \leq t \leq t_o \\ A_f(t_o)e^{-(\lambda+k_p)(t-t_o)} + \\ \dfrac{\nu \cdot C \cdot Y \cdot A_{pp}(o)}{\nu+\lambda}\left[e^{-(\lambda+k_p)(t-t_o)} - e^{-(\lambda+\nu)(t-t_o)}\right] & \end{cases} \qquad (14)$$

$$A_t(t) = A_c(t) + A_s(t) + A_g(t) + A_f(t) \qquad t \geq o \qquad (15)$$

The above equation set describes one isotope.

Once the total specific activity due to one isotope is determined by utilizing Equation 15, the predicted let-down process radiation detector response can be calculated by use of the appropriate overall detector efficiency coefficients.[8] The total predicted detector response is then determined by summing the individual responses for each isotope considered.

A computer code employing this methodology was then constructed by the author. A listing of this code is contained in Appendix A.

Results Using the Scoping Model

Using best estimate end-of-cycle values for the initial fuel pellet, initial fuel rod gap and expected primary system fission product inventories, the scoping model was run for a variety of assumed fuel failure levels.[9,10] Figure 4 illustrates the expected primary system activity levels following transients which fail one percent and one hundred percent of the fuel rods. Primary system clean-up was assumed to be re-established one hour after the transient in both of these cases. The associated predicted detector responses for the one percent and one hundred percent failed fuel cases are shown in Figure 5.

Conclusions

Based upon the results of the scoping study, the author determined that the let-down process radiation monitor's capabilities would be exceeded for the assumed fuel rod failure levels. The radiation monitor's linear response capabilities extends over a range from 10^2 to 10^8 counts per minute. Above this range, the response becomes non-linear as the saturation limit of the detector is approached. As can be seen in Figure 5, the activity levels associated with the failure of one percent or more of the fuel would overwhelm the detector. Thus, the let-down process radiation monitor could not be utilized to indicate the appropriate classification in the PVNGS Emergency Plan.

The Emergency Planning Department was informed of the author's

FIGURE 4

PRIMARY SYSTEM COOLANT SPECIFIC

ACTIVITY VERSUS TIME

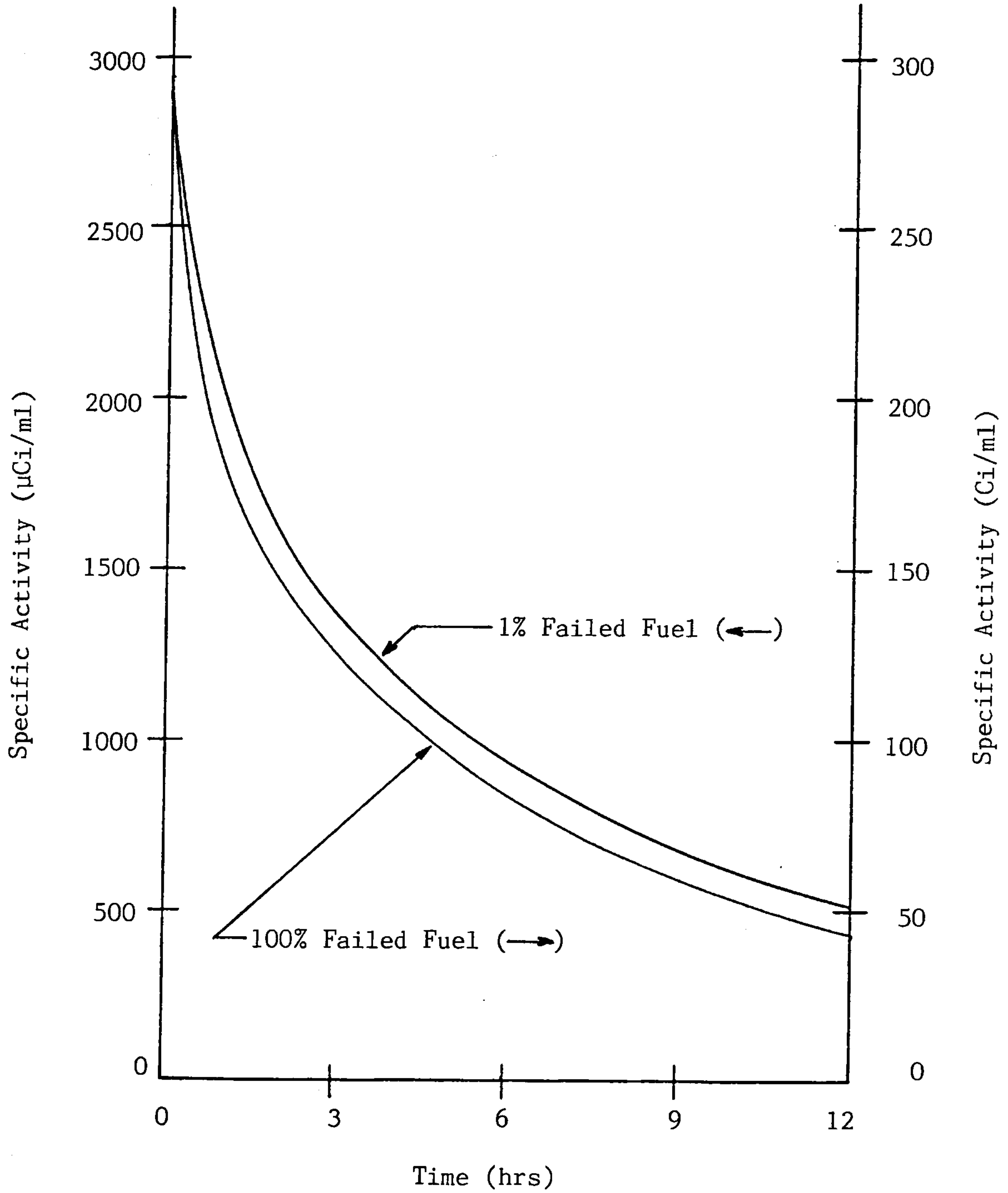
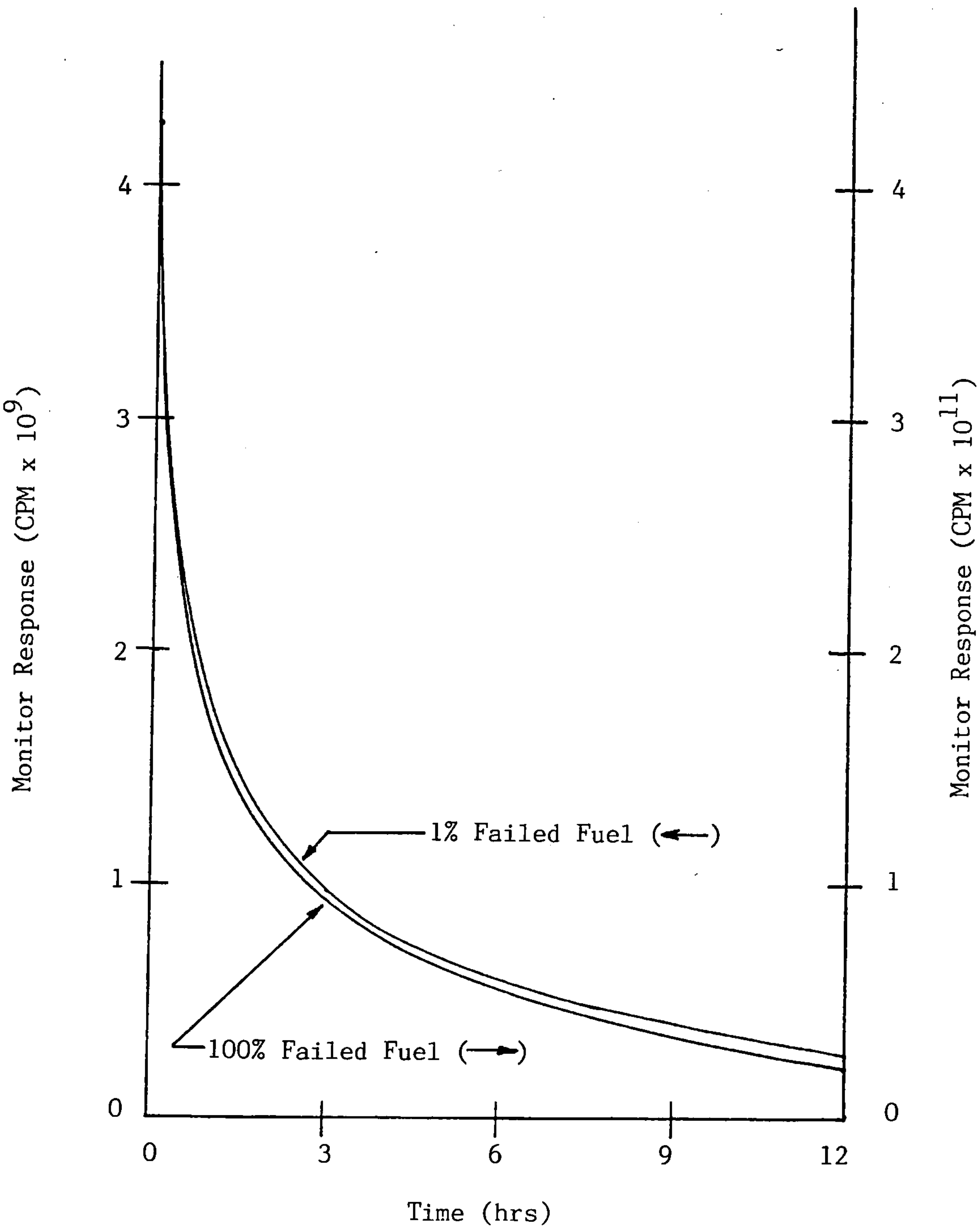


FIGURE 5

LET-DOWN LINE PROCESS RADIATION MONITOR

RESPONSE VERSUS TIME



results and conclusions based upon the use of the scoping model. It was mutually agreed upon not to pursue development of a more detailed and accurate model to correlate the fraction of failed fuel to the let-down process radiation monitor's response. Ultimately, the Emergency Planning Department relied on radio-chemical analysis of a grab sample to predict the number of failed fuel rods following an abnormal event or accident.

NUCLEAR FUEL FORECASTING SYSTEM

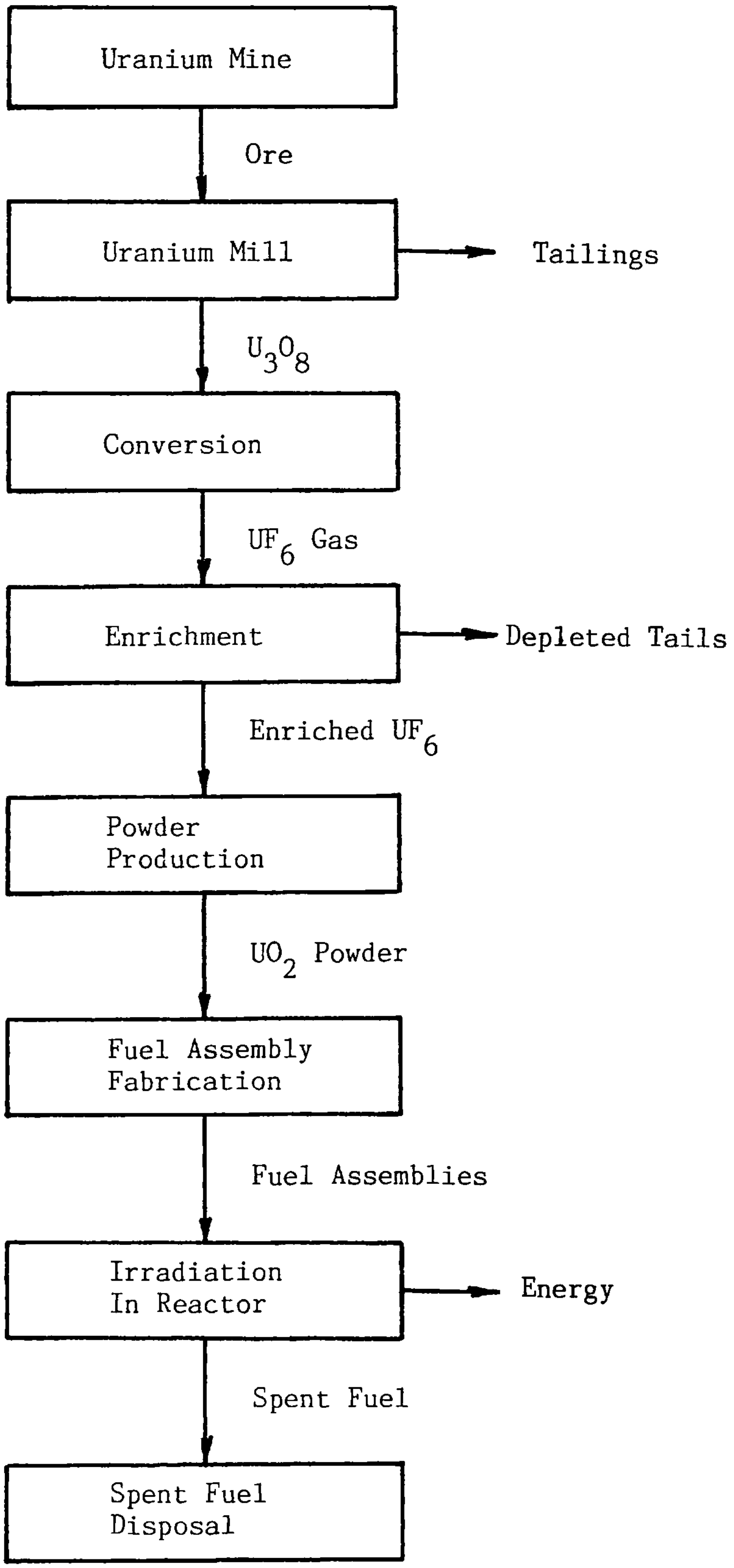
In an effort to broaden the experience of the author, the internship supervisor selected as an assignment the evaluation, selection and procurement of a nuclear fuel forecasting system. This assignment provided a valuable learning opportunity for the author for several reasons. First, the author was introduced to the procedures and the approval process associated with software evaluation and procurement. Second, the author was able to become more knowledgeable in the details of the nuclear fuel cycle. Third, the author was provided with an opportunity to become cognizant of the duties of the Fuel Cycle Services Section and the methods which are utilized to fulfill them. Fourth, the author had the opportunity to become acquainted with the many groups which Fuel Cycle Services routinely interface with. A brief description of the nuclear fuel cycle, the responsibilities of the Fuel Cycle Services Section and the author's assignment follow.

Overview of The Nuclear Fuel Cycle

The nuclear fuel cycle consists of those activities involved in procuring fabricated fuel assemblies for use in the reactor, irradiation of the fuel, as well as spent fuel disposal. Although reprocessing of the spent fuel was considered at one time and is an option in most of ANPP's contracts, reprocessing is no longer considered a viable alternative by ANPP due to political, regulatory and economic developments that have occurred during the past ten years. As such, the once-through nuclear fuel cycle is utilized for PVNGS. The basic components in this cycle are depicted in Figure 6.

FIGURE 6

ONCE-THROUGH NUCLEAR FUEL CYCLE



A brief description of the components that comprise the once-through fuel cycle follow.

The first step in the fuel cycle is the mining of uranium-bearing ores. In general, ores mined in the United States contain less than one percent uranium.[11] The ore is then chemically processed to concentrate the uranium mineral content. This process is generally performed at the mine to reduce shipping costs. The uranium concentrates are then shipped to a uranium mill for further processing and purification. At the mill, the uranium concentrates are purified by an ion exchange process and reduced to yellowcake (U_3O_8).

The U_3O_8 is then shipped to a conversion plant where the yellowcake is converted to uranium hexafluoride (UF_6). The gaseous UF_6 is then delivered to an enrichment plant. There, the U^{235} content of the uranium is increased from the naturally occurring 0.71 weight percent to between 2.0 and 4.0 weight percent for a typical fuel cycle. The exact enrichment that is required is a function of a multitude of parameters and is calculated in advance to meet the requirements of the plant. The principal method of enrichment in use in the United States today is gaseous diffusion.

The enriched UF_6 is then shipped to a powder production plant where the material is converted to uranium dioxide powder (UO_2). The UO_2 powder is shipped to a fuel assembly fabrication plant. At the fabrication plant, the powder is pressed into cylindrical pellets, sintered in a furnace to form a ceramic material and ground to final shape. The pellets are then encased in a clad tubing to form a fuel rod. The fuel rods are subsequently combined with structural components to produce a fuel assembly.

After the completion of fabrication, the fuel assemblies are shipped to the plant site for use in the reactor core. For a typical reload, between 30 and 50 percent of the fuel assemblies contained in the reactor core are replaced with freshly fabricated fuel. The reactor then generally operates for a period of time between 12 and 24 months. During this period of time, energy is extracted from the fuel assemblies through a controlled chain reaction utilizing nuclear fission.

The final step in the nuclear fuel cycle is the disposal of the spent fuel that is discharged from the reactor. Although no repository for spent fuel is currently available to operators of nuclear power plants, preparations are underway to locate, construct and operate the first repository under the direction of the U.S. Department of Energy. Until such time as the repository becomes fully operational, all spent fuel generated in the United States is stored at the plant site.

Fuel Cycle Services Section

The primary goal of the Fuel Cycle Services Section is to effectively manage the considerable present and future ANPP investment in nuclear fuel. To successfully accomplish this goal, many activities must be performed. A limited subset of these activities are:

1. Determine reload material and service requirements.
2. Evaluate the impact of contract options/amendments.
3. Forecast capital and operating budget requirements.

4. Project short-, mid- and long-range cash flow requirements.
5. Provide input to production-costing models.
6. Evaluate alternate in-core fuel management schemes.
7. Support the participants with up-to-date information and forecasts.
8. Provide information for participant rate case hearings.
9. Provide economic analysis to support the optimization of each fuel batch.
10. Develop an inventory policy for natural and enriched uranium.
11. Procure uranium, conversion services, enrichment services and spent fuel disposal services.
12. Monitor and assess the materials and services markets.
13. Process nuclear fuel allocations and invoices.
14. Plan the strategy for future fuel cycles.

To perform these and other associated tasks in a timely manner with the present and anticipated future staffing levels, a rather sophisticated software package is required.

Description of the SAROS Computer Code

When the author began his internship, Fuel Cycle Services utilized the SAROS code to perform some of the aforementioned tasks.[12] The SAROS code was developed and marketed by the S.M. Stoller Corporation. The version which ANPP utilizes, Revision 03, was obtained in August of 1978. The code is modular in design and its general computational flow is as follows:

1. Set-up input files.
2. Read input data into files.

3. Process input data.
4. Calculate specific information required for the requested output files.
5. Write the information to output files.
6. Print the requested reports.

The SAROS code requires several different types of information as input data. The first of these, fuel management scheme information, describes the reactor core, the reload batch and key operating parameters. A partial list of this type of information includes:

1. Number of fuel assemblies in the core.
2. Number of fuel assemblies in the reload batch.
3. Average initial enrichment of the reload batch.
4. Average discharge enrichment of the reload batch.
5. Weight of uranium initially contained in the reload batch.
6. Weight of uranium contained in the reload batch at discharge.
7. Initial fissile plutonium content of the reload batch.
8. Discharged fissile plutonium content of the reload batch.
9. Operating cycle length.
10. Integrated cycle energy generation.
11. Average burnups of the batches remaining in the core.

Plant operating assumptions comprise the second data set required by SAROS. This data is used to relate the batch specific timing information to actual calendar dates. To perform this task, the expected cycle capacity factors and a few specific calendar dates such as the date the unit entered commercial operation are utilized.

The third and largest data block, contract information, contains information on the price, escalation adjustments, payment schedule, delivery schedule and the losses at each stage of fuel processing. Typically, information on contracts for natural uranium (U_3O_8), conversion services, enrichment services, fuel assembly fabrication services and spent fuel disposal services are considered. Currently, ANPP has multiple contracts for each of these quantities with the exception of spent fuel disposal services.

The fourth data set contains information on market projections for the cost of each of the fuel components. This data is comprised of the escalation adjustments for the materials and services currently under contract that are expected to occur in the future. Projections of the open market prices for materials and services are also contained in the data set when no contractual coverage exists.

The fifth and final data set required by SAROS contains information concerning the expected interest rates. SAROS utilizes three separate interest rates in calculating fuel cycle costs. These are the progress payment interest rate, the working capital interest rate and the present worth interest rate. The first two of these rates are used to calculate the indirect expenses of the fuel cycle; i.e. the cost of carrying the investment in nuclear fuel over its lifetime. The third rate is utilized in all present worth calculations such as levelizing fuel cycle costs.

After the required data is read into the SAROS input files, the code performs the necessary calculations to obtain the requested output data. This portion of the SAROS code has five modules. They are:

1. Reactor Operations Module.
2. Batch Calendar Module.
3. Batch Direct Costs Module.
4. Batch Indirect Costs Module.
5. Annual and Levelized Costs Module.

The modules are executed in the sequence indicated.

The Reactor Operations Module calculates the basic quantities that are utilized by the remaining modules. These quantities enable the program to relate the reload batch to the overall reactor environment. Examples of these quantities include:

1. Relating each reload batch to the real-world calendar.
2. The fraction of the total power generated in each cycle that is assigned to a given reload batch.
3. Relating the escalation and market projection schedules to the various batches of fuel.

The Batch Calendar Module relates the various individual batch schedules, generally defined relative to the cycle start-up date, to the real-world calendar. The individual batch schedules include information on the relative timing of payments for the various components of fuel cycle, batch residency times and the time of delivery of each of the components. After execution of the first two modules, a complete schedule that contains all the significant events for a particular fuel management plan has been established.

The third module, Batch Direct Costs Module, calculates the total cost which has been incurred directly by the utility for the procurement of the reload batch. The direct cost of a reload is calculated by summing the escalated costs associated with the purchase of each of the fuel cycle components. The relative timing of each of these cash flows is not considered in determining the total direct cost of the reload batch.

The cost of carrying the considerable investment in a reload batch is calculated in the Batch Indirect Costs Module. The magnitude of the indirect costs are dependent on the timing and sequence of the payments made, credits received and amortization rate of the fuel investment. The module accounts for a variety of effects including inflation, depreciation and the possible value of any reprocessed material. Finally, the batch indirect cost and direct cost are summed to yield the total batch cost.

The fifth and final module, Annual and Levelized Costs Module, calculates the quantities that the name implies for the total reactor fuel cycle. The total annual fuel cycle cost is determined by summing the appropriate fraction of the total cost for each batch. These fractions are based upon the power generated by a batch during the given year. Finally, the annual fuel cycle cost is levelized to produce an effective cost per unit of energy generated (such as mils/KW-Hr). The calculated results are then stored in output files and the requested reports are printed.

The SAROS code had been procured for use in the Nuclear Fuel Management Department in 1978. Since the requirements for additional analyses had grown as Palo Verde neared commercial operation, this code had proven to be inadequate and of limited use. Examples of SAROS' limitations include:

1. No provision for participant ownership of Palo Verde.
2. No provision for considering inventories of natural and enriched uranium products.
3. No graphics capability.
4. No provision for multi-unit plants.
5. No provision for reinsertion of previously discharged fuel assemblies.
6. Being an extremely inflexible code with few user selected options.
7. No provision for time varying economic parameters such as interest rates, inflation rate, etc.

Efforts had been made to expand its capabilities by adding program modules. These included an increased report printing capability and improved escalation models. These additional modules had met the immediate needs of the department but had not solved the basic deficiencies of SAROS.

Preliminary Evaluation and Selection Process

Once the need to replace the SAROS code was identified, the author began a systematic study to determine the best available software system. The first step was to identify the software packages which were currently available. This was accomplished by contacting cognizant ANPP personnel, personnel from other utilities and consulting firms. A total of six software packages were identified through this process. Table 5 presents these software packages and

TABLE 5

NUCLEAR FUEL FORECASTING CODES EVALUATED

Codes	Vendor
Nuclear Fuel Forecasting System	Fuel Supply Service
FUELMACS	Pickard, Lowe, & Garrick, Inc.
Nuclear Fuel Accounting Code	NUS Corporation
Fuel Management Strategy Evaluation Code	Combustion Engineering
UFUEL	Utility Associates International
Nuclear Fuel Information System	Illinois Power Company

their respective vendors. Information was then gathered on each system and compared against a set of required features and capabilities which the author had previously established. The results of this effort are presented in Table 6.

The author then performed a qualitative evaluation of each software package utilizing the information gathered to construct Table 6. From this evaluation, two were accepted for further consideration. These were the Nuclear Fuel Forecasting System (GEM) by Fuel Supply Services (FSS) and FUELMACS by Pickard, Lowe, and Garrick (PL&G). The other four packages were determined to be unacceptable for a variety of reasons and these are briefly reviewed below.

The NUS code package was designed as an accounting tool and not a forecasting aid. As a consequence, it cannot forecast costs nor future cash flows. Also, the code cannot levelize costs to produce an effective fuel cycle cost per unit of electricity produced (i.e. mils/KWe-Hr.). Since these capabilities are an essential component of Fuel Cycle Service's needs, this package was deemed to be unacceptable.

Upon investigation, the Combustion Engineering code package was determined to be a one-dimensional reactor physics code and not a forecasting code. This code will determine enrichments, number of assemblies, cycle energy, etc. This code cannot forecast costs or cash flows nor calculate present worth or levelized fuel costs. Since this package satisfied few of the required criteria, it was determined to be unacceptable.

TABLE 6

ATTRIBUTES OF VARIOUS NUCLEAR FUEL FORECASTING CODES

System / Consideration	Fuel Supply Service NFFS	Pickard, Lowe & Garrick FUELMACS	NUS Nuclear Fuel Accounting Code	Combustion Engineering FMSEC	UAI UFUEL	Illinois Power Company NFIS
Currently in use by	Florida Power & Light	Cleveland Electric Illuminating, Southern California Edison (soon)	Gulf States Utilities & Florida Power Corp.	CE	Various Utilities	Illinois Power
Multi-Unit Capability	Yes	Yes	Yes	No	Yes	Yes
Multi-Owner Capability	Yes	Yes	Yes	No	No	Yes
By FERC Account	Yes	No	Yes			
Accounting Unit	Assembly	Batch	Assembly	Batch	Batch	Assembly
Inventory Cost Options	Weighted Average FIFO, LIFO	LIFO, FIFO, Low & High Cost Weighted (soon)	Weighted Average User Defined FIFO, LIFO		None	

TABLE 6

CONTINUED

System / Consideration	Fuel Supply Service NFFS	Pickard, Lowe & Garrick FUELMACS	NUS Nuclear Fuel Accounting Code	Combustion Engineering FMSEC	UAI UFUEL	Illinois Power Company NFIS
Amortization Report Types	By Assembly and By Component	By Batch and By Component	By Assembly and By Batch		By Batch and By Component	By Component
Modular Design	Yes	Yes	Yes		Yes	Yes
Interface Ability With NFM System	Yes	Some	Some	Yes	Some	Some
Accumulates Component Costs	Yes	No	Yes	Yes		
Accumulates Contract Costs	Yes	No	No			
IBM Compatible	Yes	Yes	Yes		No (Cybernet)	Yes, but CICS Operating System Required

TABLE 6

CONTINUED

System / Consideration	Fuel Supply Service NFFS	Pickard, Lowe & Garrick FUELMACS	NUS Nuclear Fuel Accounting Code	Combustion Engineering FMSEC	UAI UFUEL	Illinois Power Company NFIS
User Defined Escalation Clauses	Yes	Yes	No	Yes	Yes	Yes
Spent Fuel Disposal Fee	Yes	Yes	Yes			
Participant Reports	Yes	Yes	Yes	No	No	No
Calculate Costs In mil/KWH_e Or ¢/MMBTU	Yes	Yes	No	Yes	Yes	
Calculate Monthly and Yearly Costs	Yes	Yes	Yes		Yes	Yes
Reinsertion Of Bundles Option		Yes	No	Yes	Yes	

TABLE 6

CONTINUED

System / Consideration	Fuel Supply Service NFFS	Pickard, Lowe & Garrick FUELMACS	NUS Nuclear Fuel Accounting Code	Combustion Engineering FMSEC	UAI UFUEL	Illinois Power Company NFIS
Forecasts Cash Flows	Yes	Yes	No		Yes	Yes
Forecasts Costs	Yes	Yes	Yes	Yes	Yes	Yes
Accommodate Fuel Cycle Changes	Yes	Yes	Some		Yes	
Multiple Contract Sources for U_3O_8, SWU, etc.	Yes	Yes	Yes		Yes	

FERC - Federal Energy Regulatory Commission.
LIFO - Last-In - First-Out inventory accounting treatment.
FIFO - First-In - First-Out inventory accounting treatment.
SWU - Separative Work Units (enrichment services).

NOTE: A blank space indicates that the relevant piece of information was not available.

The UFUEL code was judged unacceptable primarily for two reasons. The first being the inability of the code to accommodate multiple owners of a plant. Since a significant portion of Fuel Cycle Services' time is devoted to preparing participant reports, this deficiency was deemed very significant. The second major deficiency is the incompatibility between this code which currently operates on a Cyber computer and APS' IBM computer configuration.

The Illinois Power Company's package was determined to be unacceptable primarily because it offers few advantages over the system presently utilized. It possesses the same limited capabilities and shortcomings as SAROS. Also, this package is not compatible with APS' IBM computer configuration.

Description of the Nuclear Fuel Forecasting System Computer Code

The Nuclear Fuel Forcasting System computer program, better known as the GEM code, is a modular code that performs a variety of analyses. These include fuel cycle component supply planning, financial planning, regulatory forecasting and economic decision-making. The GEM code is comprised of eleven modules with each utilizing a common data base. The input data requirements of GEM are essentially identical to those described for the SAROS code except GEM allows each of the economic parameters to vary with time. The time variance of the economic parameters permits a more realistic analysis than can be obtained with the SAROS code.

The eleven modules that form GEM are:

1. Automated File Management Module.
2. Utility Programs Module.

3. Simulation Module.
4. Requirements Module.
5. Supply Module.
6. Finance Module.
7. Inventory Cost Module.
8. Fuel Expense Module.
9. System of Accounts Module.
10. Economics Module.
11. Graphics Module.

When GEM is utilized, the modules are executed in the sequence indicated above. A brief description of each of the modules follow.

The Automated File Management Module creates the common data base needed to store the input data, calculational results and output data. The module also reads the input data. The Utility Programs Module pre-processes the input data, calculates basic quantities required by the remaining modules, initializes values of certain variables and performs a host of other similar functions.

The Simulation Module contains a two dimensional high-speed nuclear physics simulator. The module calculates the power distribution, burnup distribution, reactivity and various other parameters needed to evaluate alternate loading patterns. The Simulation Module is linked to the common data base by a self-generated card-image file. This feature allows the option to evaluate alternate physics information generated outside the GEM code such as fuel vendor supplied core designs, reference fuel management plans, etc.

The Requirements Module determines the quantity of material and services required for each reload batch. The module's calculations are based on the physics information that is calculated in the Simulation Module or supplied by the user. A schedule for the procurement and delivery of the materials and services is also constructed by the module.

The next module that is executed is the Supply Module. It determines the applicable contract price or projected market condition for each component of the fuel cycle during the period of time specified by the user. With the results from the previous module, detailed cash flows and budgets are constructed. These calculations take into account the current inventories and the utility's inventory policy.

The Finance Module calculates general economic parameters such as present worth interest rates, the cost of capital and the cost of borrowing funds. These parameters are utilized subsequently in the Economics Module and the Inventory Cost Module.

The seventh module to execute is the Inventory Cost Module. The module contains three options for treating the cost of the inventory. They are:

1. First-In-First-Out Cost (FIFO).
2. Last-In-First-Out Cost (LIFO).
3. Average Cost.

By providing these three options, GEM allows the utility to select the inventory cost policy for nuclear fuel that is consistent with its inventory cost policy for other materials.

Combining the information provided by the Supply and Inventory

Cost Modules, the Fuel Expense Module calculates the total cost of each reload batch and the cost of each of the components for each reload batch. The module then determines the total fuel cost for each cycle of operation contained in the period of interest. Finally, the module calculates the amortization rate of the fuel cycle costs.

The ninth module, System of Accounts Module, allocates the fuel cycle costs to various sets of accounts. These sets of accounts include balance sheet accounts, Federal Energy Regulatory Commission (FERC) accounts and user defined accounts. The module also determines the allocation of the fuel cycle costs to each of the participants.

The Economics Module utilizes information from many of the other modules to calculate such quantities as present worth, rate of return, revenue requirements, discounted cash flows and levelized fuel costs. The Economics Module is extremely flexible in nature thus allowing the user to perform a multitude of economic analyses. GEM also allows the Economic Module to perform analyses on data generated outside of the code. Thus, the utility of the GEM code is further enhanced.

The final module, the Graphics Module, provides the capability of outputing the calculational results of GEM in a variety of formats. The possible formats include:

1. Bar charts.
2. Pie charts.
3. Single and multiple line graphs.
4. Tables.
5. Reports.
6. User defined.

The module's capabilities provide a quick and easy to use method of preparing the final product of an analysis, the report.

Final Selection Process

Subsequent to the preliminary evaluation, both FSS and PL&G made presentations and provided detailed documentation of their respective codes. At the author's request, each also prepared and submitted a proposal for consideration. After considering these two proposals, the author concluded that ANPP should purchase the software package and related options from FSS. Since both packages met the detailed criteria specified in Table 6, the final selection was based upon several overall considerations. One of the most important being that the FSS system is a much more "mature" and proven product with several years of use by an operating utility in situations very similar to ANPP. Because of this, the costs associated with customizing the software to ANPP's particular set of circumstances should be minimal if not zero. In comparison, the PL&G package has never been utilized at an operating utility. Past experience with unproven codes indicate that significant levels of resources will be required to make the code useable at ANPP.

Also, the FSS system has additional capabilities that far exceed those of its competitor. Examples of these include a more sophisticated graphics package and a built-in two dimensional physics simulator. Although not a current requirement, the physics simulator will allow Fuel Cycle Services to perform reload optimization studies. This capability will become very important when Nuclear Fuel Management begins to do reload core design. These and other additional capabilities will enable Fuel Cycle Services to perform more effectively and efficiently now and into the future. The final consideration was that the FSS system is slightly less expensive than the PL&G package.

After completion of the final selection process, the author prepared a report delineating the selection process and its eventual outcome. A recommendation and supporting justification was also prepared by the author. After the review of this report by ANPP upper management, the author's recommendation was accepted and negotiations were begun with FSS. By the end of the internship period, ANPP's upper management had reached an agreement in principle with Fuel Supply Services for the procurement of their code package. Subsequent negotiations were required to reach a concensus on a software license agreement. The contract for the purchase of the Nuclear Fuel Forecasting System was ultimately executed in September of 1985.

THERMAL MARGIN IMPROVEMENT PROGRAM NEGOTIATIONS

Through detailed thermal-hydraulic analyses and the results from the pre-core Hot Functional Flow Test, it was evident that Palo Verde had insufficient thermal margin for effective full power operation. The lack of thermal margin was due to a number of identifiable causes. These are:

1. The actual performance of some Palo Verde systems do not meet the original design criteria. Examples of inadequate system response include the performance of the High and Low Pressure Safety Injection Systems.
2. During the construction and start-up phases, Palo Verde was subjected to a number of NRC imposed penalties and new requirements. Examples include new statistical treatment of the critical heat flux experimental data and a penalty due to the uncertainty of the effects of spacer grids on the critical heat flux experimental data.
3. ANPP elected to operate Palo Verde on 18 month cycles as opposed to the originally envisioned annual cycles.

To remedy this undesirable situation, Combustion Engineering proposed a Thermal Margin Improvement Program to ANPP.[13,14] After evaluating the merits of the proposal, ANPP determined that it was technically adequate but that the terms on cost responsibility were not equitable. The author was given the task to research the complex legal and engineering aspects of the issue, to prepare a report which would provide the bases for subsequent negotiations with Combustion Engineering, and to present the results of this effort to management. A brief description of the Core Protection Calculators, the Core Operating Limits Supervisory System, the Thermal Margin Improvement Program and its benefits, the results of the author's research, and the outcome of the negotiations with Combustion Engineering are detailed below.

Core Protection Calculators

The Core Protection Calculators (CPCs) are digital computers and their associated software which are contained in the Palo Verde Reactor Protection System. The overall function of the CPCs is to assure that Specified Acceptable Fuel Design Limits are not exceeded during anticipated operational occurrences. As used here, the term anticipated operational occurrences is defined as those conditions of normal operation and transients which are expected to occur one or more times during the life of the power plant. Particular examples of these occurrences include loss of power, dropped control element assembly (CEA), single failure of an electrical component, failure of a control system, sheared reactor coolant pump shaft, and loss of main feedwater to the steam generators. The CPCs are also designed such that reactor shutdown (trip) is not initiated during normal operations.

The Reactor Protection System consists of four independent measurement and protection channels, hence, there are four CPCs. The four CPC channels provide trip signals to a two-out-of-four or a two-out-of-three coincidence logic. This redundancy allows the necessary protection to be achieved while allowing for one channel to be taken out of service for maintenance, testing or calibration.

The associated output signals and input signals, with the exception of the Control Element Assembly Calculators (CEACs), are also electrically and physically separated for each channel. There are only two CEACs which provide information to all four CPC channels on the position of the CEAs.

The CPCs are specifically designed to ensure that two Specified Acceptable Fuel Design Limits, fuel centerline melting and departure from nucleate boiling, are not exceeded. To accomplish this task, two parameters are calculated by the CPCs from the input signals and are compared against fixed, preset values. These are the peak local power density and the minimum departure from nucleate boiling ratio (DNBR). If the calculated values are less conservative than the preset values, a trip signal is generated by the CPCs.

The inputs utilized in the calculation of local power density and DNBR are detailed in Table 7. These input signals are digitized and conditioned by multiplexing and analog-to-digital conversion equipment which is part of the calculator hardware. The following calculations are performed by the CPCs or CEACs:

1. CEA group deviations (misalignment of individual CEAs within a group).
2. Correction of ex-core flux power for shape annealing and CEA shadowing.
3. Reactor coolant flow rate.
4. Core average thermal power from reactor coolant temperature and flow rate information.

TABLE 7

CORE PROTECTION CALCULATOR INPUTS

Signal	Number Per Channel
Core inlet temperature	4
Core outlet temperature	2
Pressurizer pressure	1
Reactor coolant pump speed	4
Ex-core detector signal	4*
CEA position	

*Each ex-core detector has three independent sections.

5. Calibration of the ex-core flux power to the core average thermal power.
6. Axial power distribution from the corrected ex-core flux power signals.
7. Fuel rod and coolant channel planar radial peaking factors.
8. Departure from nucleate boiling ratio.
9. Comparison of DNBR with a fixed trip setpoint.
10. Peak local power density based upon the existing power distribution.
11. Comparison of calculated peak local power density with a fixed trip setpoint.
12. Determine if a CEA group deviation alarm is required.

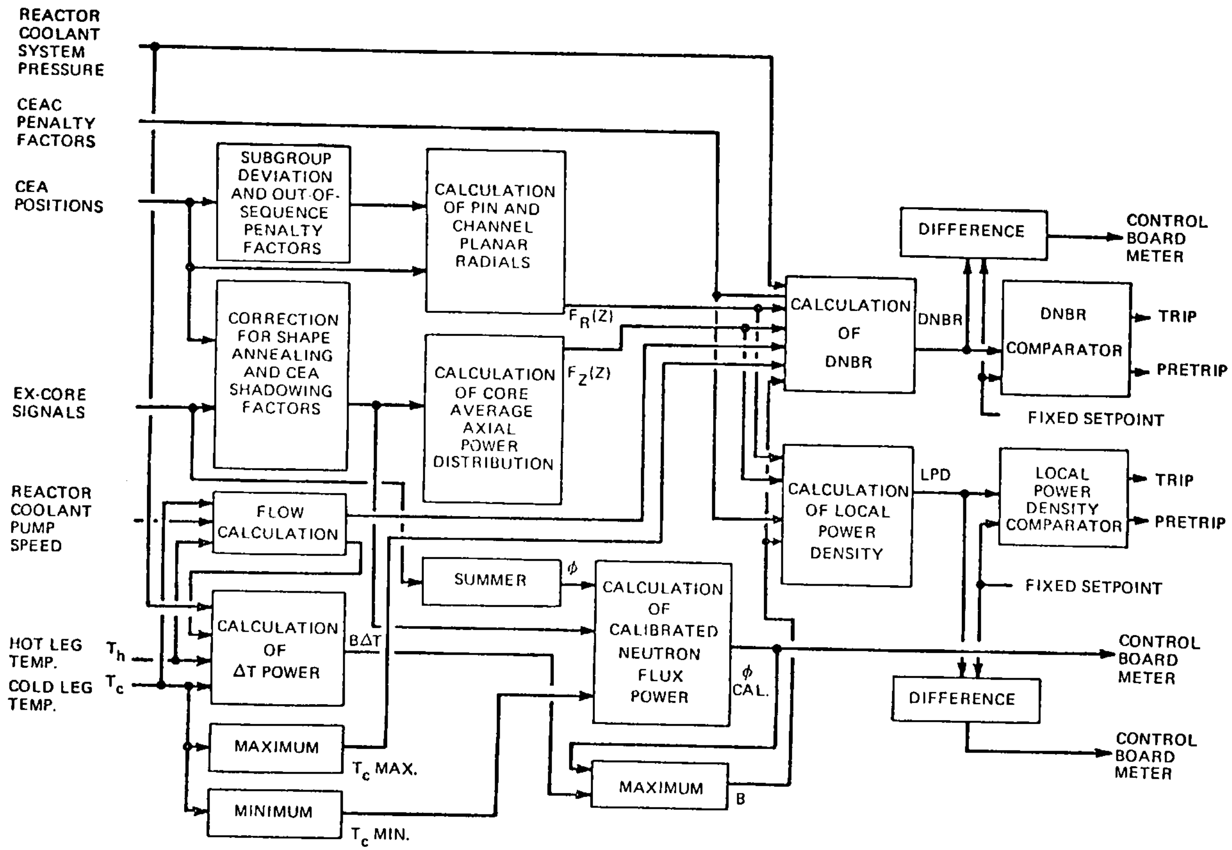
A simplified flow diagram of the calculations performed by the CPCs is illustrated in Figure 7. The outputs which the CPCs generate are shown in Table 8.

Core Operating Limits Supervisory System

The Core Operating Limits Supervisory System (COLSS) consists of process instrumentation, algorithms and operator displays which continually monitor important plant parameters. The purpose of COLSS is to monitor and provide information on reactor core conditions and ensure that they are no more severe than is permitted by the Limiting Conditions for Operation. The PVNGS Technical Specifications define the Limiting Conditions for Operation within which the plant can operate without violating its license. The values of the Limiting Conditions for Operation are defined such that the reactor core conditions during operation are no more severe than the initial

FIGURE 7

CORE PROTECTION CALCULATOR FUNCTIONAL BLOCK DIAGRAM*



*Taken from Reference 10

TABLE 8

CORE PROTECTION CALCULATOR OUTPUTS

DNBR Pretrip Alarm

DNBR Trip Signal

Peak Local Power Density Pretrip Alarm

Peak Local Power Density Trip Signal

DNBR Margin*

Local Power Density Margin*

Calibrated Ex-core Flux Power*

Control Element Assembly Withdrawal Prohibit

*Signals utilized for control room indication (meters).

conditions assumed in the safety analyses and in the design of the CPCs.

Simplistically, COLSS calculates two important parameters - margin to a limiting core power and azimuthal tilt. The margin to a limiting core power is based upon DNBR limits, peak local power density limits and licensed power limits. The azimuthal tilt is synthesized from the network of in-core self-powered rhodium detectors. In calculating these quantities, the input signals listed in Table 9 are utilized. These signals are conditioned and digitized before becoming input to the COLSS algorithms.

The following parameters are calculated by COLSS:

1. The reactor coolant volumetric flowrate.
2. The reactor core power based upon core inlet temperature, outlet temperature and coolant flowrate.
3. The reactor core power based upon a secondary system calorimetric measurement.
4. The reactor core power based upon the turbine first stage pressure.
5. The peak local power density power operating limit.
6. The DNB power operating limit.
7. The margin to the peak local power density power operating limit.
8. The margin to the DNB power operating limit.
9. The margin to the licensed core power.

Numerous other less important parameters are also calculated and/or monitored by COLSS to assist the plant operators. A simplified

TABLE 9

CORE OPERATING LIMITS SUPERVISORY SYSTEM INPUTS

Signal	Number of Sensors
Reactor coolant pump rotational speed	2 per pump
Reactor coolant pump differential pressure	2 per pump
Cold leg temperature	1 per cold leg
Hot leg temperature	1 per hot leg
Steam generator feedwater flow	1 per generator
Steam flow	1 per generator
Steam generator feedwater temperature	1 per generator
Steam pressure	1 per generator
In-core detector system	61 in-core assemblies each containing 5 axially spaced detectors
CEA position	1 per CEA
Pressurizer pressure	2
Turbine loop pipe pressure	2

block diagram of COLSS is depicted in Figure 8. The outputs which COLSS drives are delineated in Table 10.

Thermal Margin Improvement Program Description and Benefits

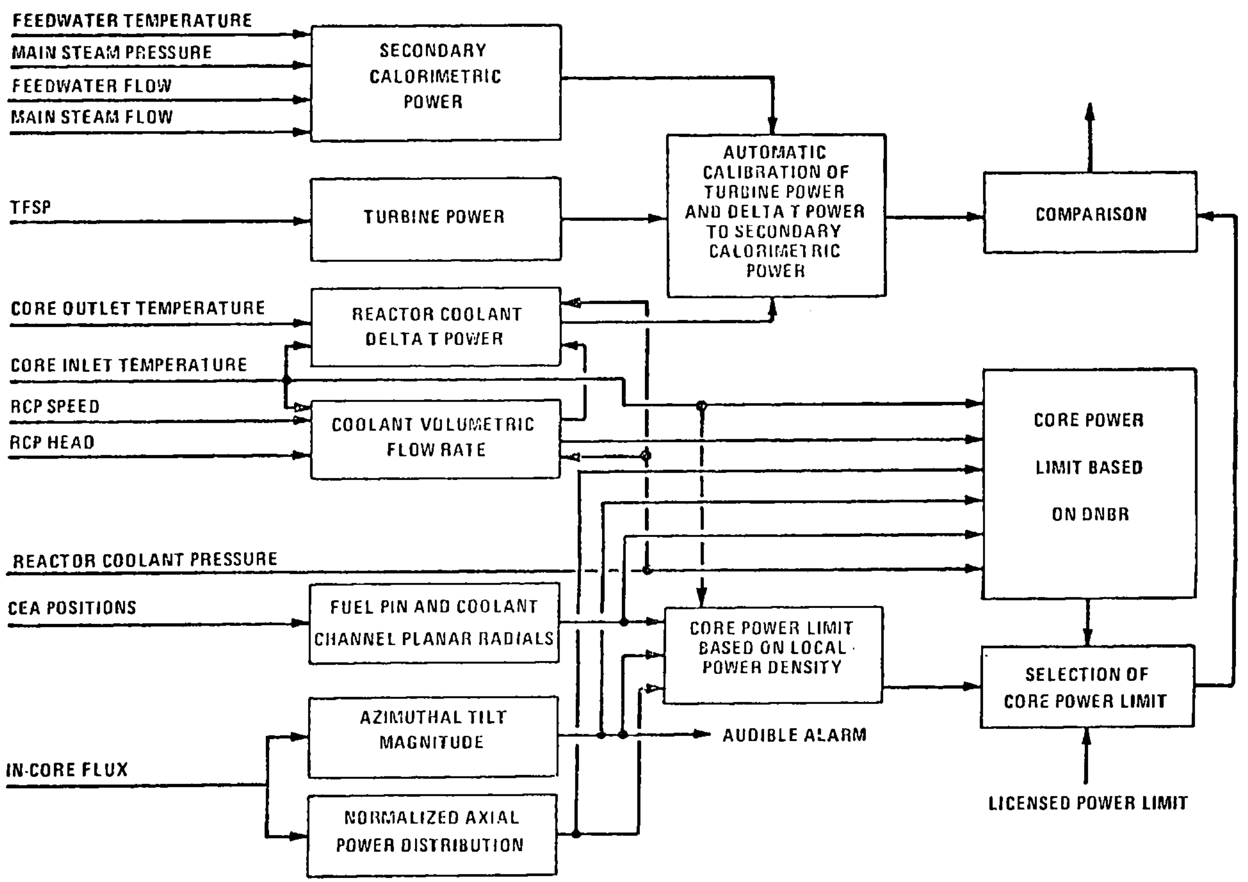
At ANPP's request, CE performed best estimate analyses to determine Palo Verde's thermal margin. The results of these analyses, thermal margin versus time, are depicted in Figures 9 and 10 for 12 month and 18 month later cycles, respectively. The quantity shown in both of these figures represents thermal margin to the Core Protection Calculator (CPC) pre-trip alarm. The actual CPC trip signal is generally generated at a power level three percent of full power greater than the pre-trip alarm.

As used here, the term "thermal margin" represents the more limiting value of CPC thermal margin and the Core Operating Limits Supervisory System (COLSS) thermal margin. CPC thermal margin is equal to the difference between the licensed maximum power level and the power level which, if attained, would induce a CPC trip. This difference is generally expressed in percent of full power. Similarly, COLSS thermal margin is equal to the difference between the licensed maximum power level and the COLSS calculated Power Operating Limit. At present, the CPC thermal margin is the more restrictive of the two for Palo Verde.

The program which CE proposed consisted of additional engineering analyses and computer software algorithm changes to attain an increase in the thermal margin. The software for the CPC and COLSS would be modified to provide a more accurate (less conservative) calculation of DNBR and peak linear heat rate. A summary of the program and its

FIGURE 8

CORE OPERATING LIMITS SUPERVISORY SYSTEM FUNCTIONAL BLOCK DIAGRAM*



*Taken from Reference 10

TABLE 10

CORE OPERATING LIMITS SUPERVISORY SYSTEM OUTPUTS

Core power operating limit based on peak local power density

Core power operating limit based on margin to DNB

Total core power

Margin between core power and nearest core power operating limit

Axial shape index

Azimuthal tilt

Numerous alarms

Numerous reports are available via a teletype

FIGURE 9

THERMAL MARGIN VERSUS TIME FOR

TWELVE MONTH LATER CYCLES

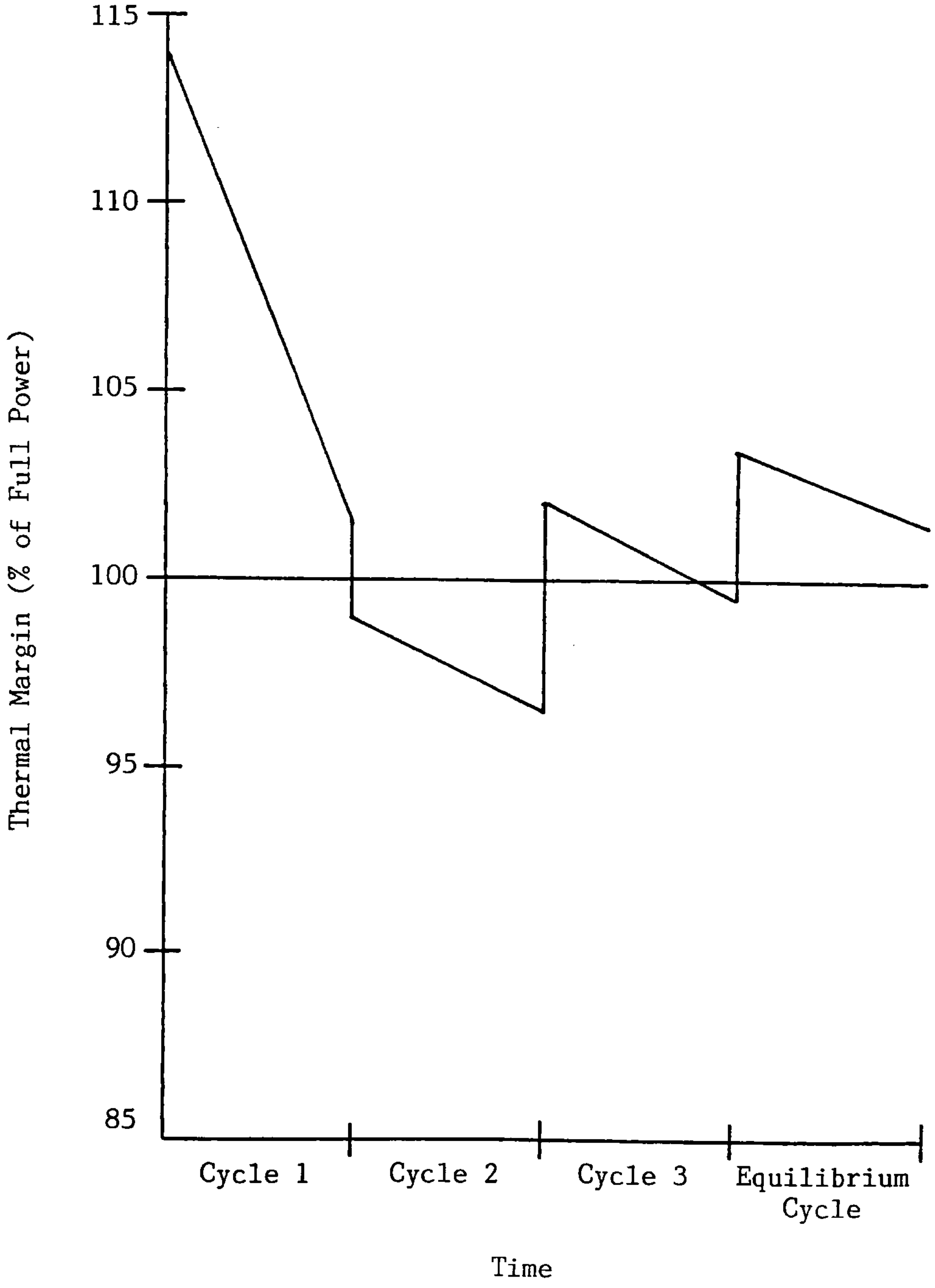
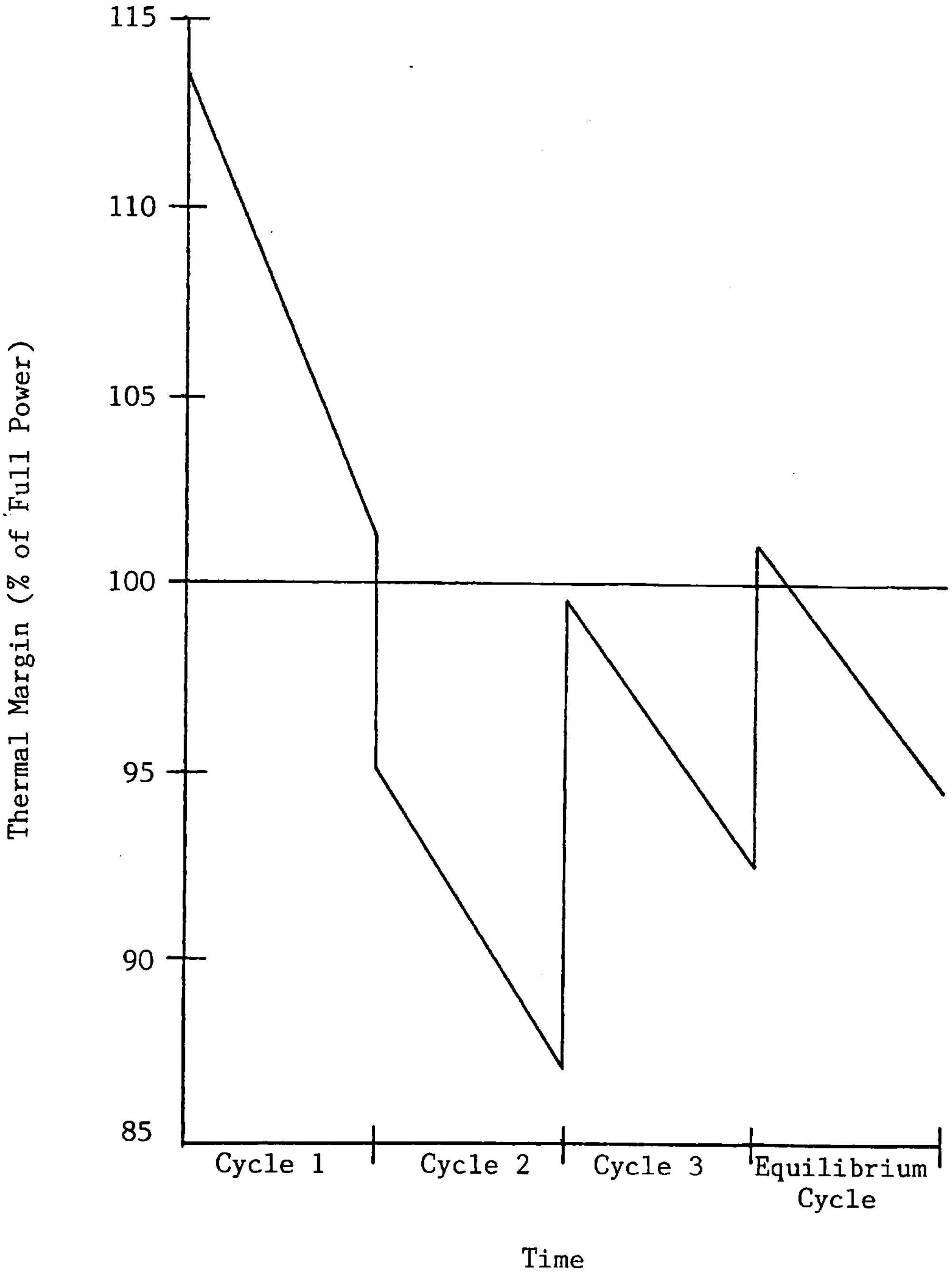


FIGURE 10

THERMAL MARGIN VERSUS TIME FOR

EIGHTEEN MONTH LATER CYCLES



benefits are provided in Table 11. The projected margin gains are also illustrated in Figure 11 for 18 month later cycles. A more detailed description of the components of the Thermal Margin Improvement Program follows.

As depicted in Table 11, the Thermal Margin Improvement Program is comprised of eleven components. Each of these components is either a modification of the algorithms in COLSS or the CPCs or a change in the analysis methodology which is used in selecting the appropriate constants for installation in COLSS or the CPCs. A brief description of each of these components follow.

Density Dependent F_{xy}

The present COLSS algorithm does not allow for adjustment of the radial peaking factor (F_{xy}) values for variations in the inlet moderator density due to temperature. The present algorithm utilizes a value which is always conservative when the inlet temperature varies within the Limiting Conditions for Operation. The effect of utilizing this conversative value is that it penalizes the COLSS system during normal operation.

With the implementation of this program, the COLSS algorithm would be modified to give the radial peaking factor a slight dependence on inlet moderator density. In this manner, the radial peaking factor is reduced when operating under nominal conditions but still retains sufficient conservatism for off-nominal operating conditions. An increase of approximately 3.0 percent in COLSS thermal margin is projected with the implementation of this component.

TABLE 11

COMPONENTS OF THE THERMAL MARGIN IMPROVEMENT

PROGRAM AND THEIR ASSOCIATED BENEFITS

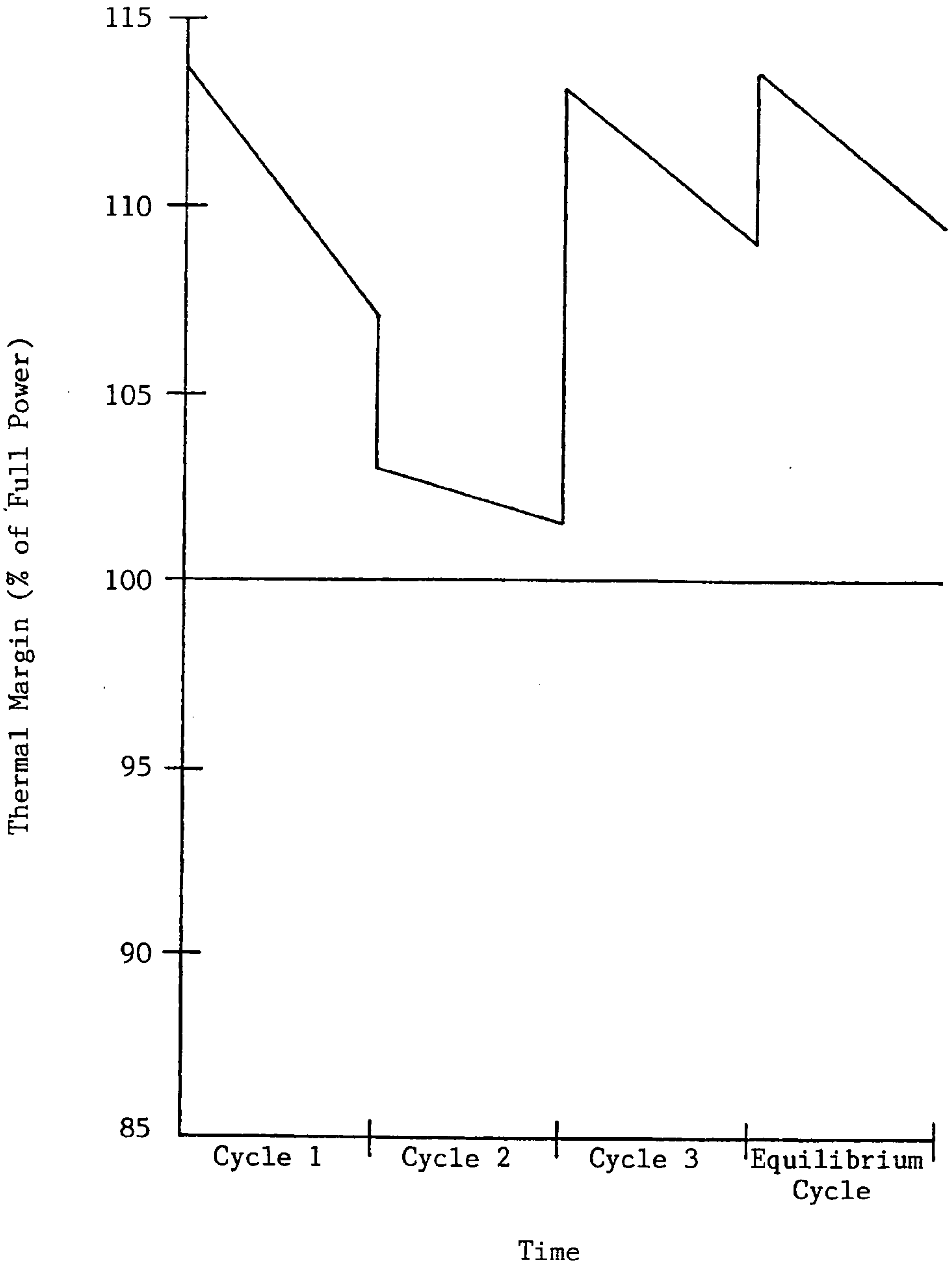
		ESTIMATED IMPROVEMENT*		
		COLSS	CPC	
			12 Mo.	18 Mo.
1.	Density Dependent F_{xy}	3.0%	-	-
2.	Aximuthal Tilt Algorithm Improvement	1.0%	0.5%	1.0%
3.	UPDATE Algorithm Improvement	-	2.0%	2.0%
4.	Extended SCU Method (System Parameters/State Parameters)	**	1.5%	1.5%
5.	Power Uncertainty as a Function of Power Level (BERR0, BERR4)	-	3.0%	3.0%
6.	Power Distribution Algorithm Improvement (BOC, EOC)	-	1.0%	3.0%
7.	Two-Region State Parameters	0.5%	1.0%	1.0%
8.	Dynamic Compensation Penalty Factor Reduction	-	1.5%	1.5%
9.	Statistical Transient Analysis (CEOG)	5.0%	-	-
10.	Burnup Dependent BERR1, EPOL2	1.0%	0.5%	1.5%
11.	Partial Elimination of F_{xy} Uncertainty on DNB	1.7%	1.0%	1.0%
	TOTAL	12.8%	12.6%	16.6%

*CPC column estimates margin improvement for both 12 and 18 month second cycles.

**Margin gain included in item 9.

FIGURE 11

THERMAL MARGIN VERSUS TIME FOR EIGHTEEN MONTH LATER CYCLES AFTER IMPLEMENTATION OF THE THERMAL MARGIN IMPROVEMENT PROGRAM



Azimuthal Tilt Algorithm Improvement

The COLSS algorithm utilizes 61 strings of self-power rhodium detectors to monitor the core power distribution. Each of these strings is comprised of five axially spaced individual detectors. For the purposes of computing azimuthal tilt, the 61 strings are assigned to one of nine groups. Thus, there are 45 separate indications of tilt - nine groups, five axial levels. The five axial indications in each group are arithmetically averaged to yield an estimate of core average azimuthal tilt. The current COLSS software utilizes the maximum of these nine values in its calculations. In this manner, the core average azimuthal tilt is always conservatively overestimated.

The modified algorithm will use a vector averaging technique for calculating the azimuthal tilt. The five axial indications of tilt in each group will be averaged vectorially. The largest of the nine averaged tilts will then be utilized as the estimate of core average azimuthal tilt. By introducing the vector averaging technique, the components of the individual tilt estimates that are due to system noise will effectively offset each other. In general, the vector averaged tilt estimate will always be smaller than the arithmetically averaged tilt estimate. This component of the program should yield an increase of approximately 1.0 percent in COLSS thermal margin.

Since the CPCs employ the ex-core flux detectors instead of the in-core rhodium detectors, an estimate of tilt must be manually entered into the software. By Technical Specification, the CPC tilt estimate must be greater than the azimuthal tilt calculated by COLSS.

By decreasing the COLSS calculated tilt with this new algorithm, the CPC tilt estimate can be decreased. A 1.0 percent gain in CPC thermal margin is also projected with the implementation of this component (assuming 18 month later cycles).

UPDATE Algorithm Improvement

In the present CPC software, a detailed calculation of DNBR is performed in the STATIC subroutine which executes every two seconds. To ensure conservative predictions under rapidly changing plant conditions such as those expected during postulated accidents, the DNBR calculated in STATIC is adjusted every 50 milliseconds by the UPDATE subroutine. This is accomplished by the use of simple derivatives of DNBR with respect to various plant parameters such as core power, inlet temperature, core flow rate, etc. To ensure conservatism, a constant penalty is always applied by UPDATE when it adjusts the STATIC calculated DNBR. The value of this penalty is sufficiently large that the estimate of DNBR is always conservative under all possible operating conditions.

The change in the UPDATE subroutine will replace the uniform penalty with a three-level one. If the plant conditions have not significantly changed from those used in the STATIC calculation, no penalty would be applied. If the plant was operating at or near nominal operating conditions, only a small penalty would be assessed. When plant conditions are off-nominal, a larger penalty would be applied to account for increased sensor errors. This modification will increase the CPC thermal margin approximately 2.0 percent when implemented (assuming 18 month later cycles).

Extended SCU Method

The current CPC software utilizes a fixed penalty factor applied to the calculated DNBR which accounts for both the system and state parameter uncertainties. System uncertainties include both computational and methodology uncertainties while state parameter uncertainty includes measurement and signal processing uncertainties. The present fixed value penalty factor is calculated with the Statistical Combination of Uncertainties (SCU) methodology which, as the name implies, combines the uncertainty components statistically.

With the implementation of the extended SCU methodology, the overly conservative fixed value penalty factor will be replaced with a probability distribution function which describes the combined system and state parameter uncertainties. The probability distribution function will then be utilized in calculating the DNBR at the 95 percent probability, 95 percent confidence level. This component of the program will increase CPC thermal margin approximately 1.5 percent for 18 month later cycles.

Power Uncertainty As A Function Of Power Level

Present methodology employs single, limiting values for penalty factor constants which are used to adjust the heat flux, local power density, thermal power and neutron flux power level in the CPCs. These penalty factors account for such things as system and state parameter uncertainties, radiation induced fuel rod bow, variation in the fabrication of the fuel assemblies, and many others. The values chosen for the penalty factors are such that conservative values for the heat flux, local power density, thermal power and flux power are calculated for all anticipated operating and accident conditions.

The new CPC software will utilize penalty factors which are functions of core power level. This is expected to produce a penalty factor of the same magnitude at low power levels as the present, fixed penalty factor but should provide a substantial reduction in the penalty factors at full power. This results from generally lower instrumentation errors and corresponding higher signal-to-noise ratios at full power. This component to the program is expected to yield a 3.0 percent increase in CPC thermal margin assuming 18 month later cycles.

Power Distribution Algorithm Improvement

The CPC power distribution algorithm synthesizes the core average axial power shape based on the multi-level ex-core neutron detector responses. An important step in this process is the selection of an appropriate set of cubic spline functions and the determination of their respective amplitudes which best characterize the multi-level detector responses. The present CPC software contains seven sets of spline functions but due to changes which have occurred since the initial design of the system, a single set of spline functions is always selected for use. This condition generally leads to a poorer fit to the measured power shape and, therefore, leads to higher uncertainties in the power sythnesis algorithms.

As part of the program, the number of available sets of spline functions will be expanded to fifteen and their respective shapes will be revised to more closely match the power shapes which are expected during operation. In this manner, the synethesized power shape should agree more closely to the measured power shape. Implementing this component of the program should increase CPC thermal margin by 3.0 percent assuming 18 month later cycles.

Two-Region State Parameters

In the present methodology, the thermal-hydraulic and DNBR overall uncertainty analysis is performed over the region of possible operating space which is defined by the Limiting Conditions for Operation and/or the Limiting Safety System Setpoint boundaries for the CPCs and COLSS. This analysis is used to determine the fixed penalty factors which, when applied, produce conservative CPC calculated values of minimum DNBR and maximum LPD. The COLSS power operating limit penalty factor is similarly affected by the results of the uncertainty analysis.

When the program's new methodology is implemented, the overall uncertainty analysis will be performed over two regions of operating space instead of one. The first would be for near nominal operating conditions while the second would encompass the remaining operating space contained within the present boundaries. The effect of this change will be that small penalty factors will be applied in the near nominal operating conditions region. The penalty factors will, therefore, be region dependent as well as power dependent as previously described in the Power Uncertainty As A Function Of Core Power Level component. The projected benefit of this component of the program is 0.5 percent COLSS thermal margin and 1.0 percent CPC thermal margin with 18 month later cycles.

Dynamic Compensation Penalty Factor Reduction

One of the penalty factors utilized by the CPCs is used to explicitly account for non-conservatism in the CPC calculation of thermal power and reactor vessel inlet temperature during extremely rapid transients such as control element assembly ejection. This offset provided by the penalty factor accounts for the lag in dynamic response of the CPCs caused primarily by the relatively long temperature sensor response time and the periodic execution of the CPC algorithms.

The new methodology will attempt to reduce the magnitude of this penalty factor by a variety of analytical improvements. These include new CPC algorithms, new methods of determining bias, and improved benchmarking of the CPC results. A 1.5 percent increase in the CPC thermal margin is expected with the implementation of this component assuming 18 month later cycles.

Statistical Transient Analysis

The constants which are installed in the COLSS algorithms are determined to some extent by performing transient analysis for a variety of accidents and anticipated operational occurrences. These transient analyses are performed in a deterministic manner, that is, the worst initial conditions, system operation, uncertainties, etc. are assumed. In this manner, it is assured that conservative results are calculated. This in turn assures that COLSS provides conservative results.

With the implementation of the program, a more statistical approach to performing transient analysis will be utilized. An approach similar to that of the SCU methodology will be used. The input to the analyses will be varied instead of assuming they are all in the most adverse condition. The transients will then be analyzed to achieve results at the 95 percent probability level and 95 percent confidence level. A 5.0 percent increase in COLSS thermal margin is effected with this change.

Burnup Dependent Penalty Factors

At present, the overall uncertainty analysis used to set the penalty factors only consider the worst point in the cycle burnup. As a consequence, conservative values must be utilized for the penalty factors to account for the expected variations due to burnup. This effect impacts both the CPC and COLSS thermal margins in a similar manner.

The new methodology will allow the installation of different sets of penalty factors over the course of the cycle. This will tend to reduce the penalty factors near the end of cycle when, in general, less thermal margin exists. It is projected that a 1.0 percent increase in COLSS thermal margin and a 1.5 percent increase in CPC thermal margin (assuming 18 month later cycles) will be achieved with this component.

Partial Elimination of F_{xy} Uncertainty On DNB

The present methodology incorporates a single, fixed penalty factor to account for uncertainties on the planar radial peaking factors, F_{xy}, at all axial levels. This methodology does not distinguish differences in F_{xy} uncertainties at the five axial levels measured by the in-core neutron detectors. This methodology also does not distinguish between uncertainties due to random noise and other types.

The new methodology will account for the varying uncertainty on F_{xy} as a function of core axial location. The benefit to thermal margin is due primarily to being able to statistically combine the random component of the uncertainty at each axial location as the code integrates up the coolant channel to the point of minimum DNBR. This component is expected to produce an increase of 1.7 percent in COLSS thermal margin and 1.0 percent increase in CPC thermal margin for 18 month later cycles.

As Figure 11 shows, the proposed program would provide sufficient thermal margin to allow effective full power operation at the end of Cycle 2. In Cycle 3 and beyond, excess thermal margin, above that required for reliable full power operation, would be available. This excess margin could be used for:

1. Increased operational flexibility.
2. Attaining stretch power rating.
3. Increased power capability for COLSS out-of-service conditions.
4. Increased fuel management flexibility such as long cycles, low leakage fuel management, axial blankets, coast down, etc.
5. Increased plant availability through a greater ability to withstand expected transients.
6. Mitigating the consequences of future NRC licensing requirements.

7. Mitigating the consequences of plant equipment degradation such as increased steam generator tube plugging.

Preparations for the Negotiations

The author was assigned the task of developing and defining a position from which ANPP could negotiate in the upcoming discussions with CE. Due to the magnitude of the cost associated with the program and the necessity of purchasing it, the author spent a considerable portion of the internship on this project. The author began this task by researching the Nuclear Steam Supply System Contract, the CE Fuel Contract, the pre-award contract bid specifications, the evolution of new post-award licensing requirements, as well as any applicable correspondence during the pertinent time period. From this research, the author constructed the chronology of the events that precipitated the current thermal margin problem. Also, the author developed an understanding of the complex relationship between the decisions that were made and their respective impact on thermal margin.

The information gained from the author's research activities was augmented by a number of interviews. Individuals on the pre-award contract bid evaluation team, representatives from management and technical experts in the various areas of contention were contacted by the author. These individuals included the ANPP Project Director, the Manager of Nuclear Engineering, the Manager of Licensing, the Assistant Vice President for Nuclear Production and the Supervisor of the Safety Analysis Section. From these interviews, the author was able to place the documents previously researched into the proper perspective. The author was also able to establish the intent of the parties at the time the various contracts were signed.

After the research into the pertinent issues was completed, the author prepared an outline describing the legal basis supporting ANPP's position. The author then met with ANPP's legal counsel on several occasions to discuss and more fully develop the legal basis. From these interactions with the legal counsel, the author gained insights into and an appreciation for the fundamentals of contract law.

The culmination of this effort was the preparation of a paper by the author delineating a negotiating position and providing justification for it. This report was subsequently reviewed by various levels of management, ANPP's legal counsel and various technical groups. After incorporation of all appropriate comments, the report was issued to management and the negotiating team.[15]

<u>Result of the Negotiations</u>

Using the report as the basis for ANPP's arguments, the author and the other members of the negotiating team met several times with CE to discuss the issue of thermal margin. CE ultimately conceded that the arguments being forwarded by ANPP were essentially correct. An equitable settlement for the purchase of the Thermal Margin Improvement Program was obtained between CE and ANPP on November 15, 1985.[16,17] The resulting settlement required ANPP to purchase the entire Thermal Margin Improvement Program for approximately one-half of its original cost. The portion of the program that Combustion Engineering performed at no cost to ANPP recovered sufficient thermal margin to meet its contractual obligations.

CONCLUSIONS

The author's internship with the Arizona Nuclear Power Project was both a rewarding and a highly educational experience. The internship provided an unique opportunity for the author to gain valuable insights and to make identifiable contributions to such diverse technical areas as power plant operations, nuclear fuel cost accounting and forecasting, reload planning, plant design features, licensing, quality assurance and emergency planning. Through the three primary tasks described in this report and numerous less substantial assignments, the author was provided an opportunity to apply the knowledge previously gained in the academic portion of the Doctor of Engineering program.

The internship also allowed the author to gain valuable experience in a wide range of non-technical areas. These included such diverse areas as contract administration, procurement, contract negotiations, contract law, accounting, budget preparation and public speaking. By combining both technical and non-technical aspects in the assignments, ANPP provided the author with a well-rounded educational experience which the author can build upon in the future.

The most important lesson which the author learned during the internship was the realization that the engineer must be able to effectively communicate. Without this ability, much of the technical value of an engineer's work is lost. During the internship, the author strived to develop both his verbal and written communication skills. In this manner, the value of the author's work to the company was enhanced.

The successful completion of the internship objectives was due in part to the efforts of both the internship supervisor, Dr. Wm. Bruce Miller, and the Manager of Nuclear Fuel Management, Mr. Paul F. Crawley. Both took an active role in the internship by providing direction of the author's activities and by assigning appropriate tasks to the author. Through these tasks, the author was able to gain valuable experience as a practicing engineer while contributing to the overall success of the Arizona Nuclear Power Project.

Overall, the author's intern experience was a success. The internship provided an opportunity for the author to learn many valuable lessons as well as satisfy the requirements of the Doctor of Engineering program. The experience will provide a solid foundation for future endeavors by the author.

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APPENDIX A

The following pages are the listing of the Fuel Failure Correlation Computer Code.

```
10 REM    THIS PROGRAM CALCULATES THE ACTIVITY AND LET-DOWN RADIATION MONITOR RESPONSE
20 REM    AFTER A TRANSIENT. THE ACTIVITY IN THE PRIMARY SYSTEM IS ASSUMED TO COME
30 REM    FROM FOUR SOURCES: 1) NORMAL COOLANT ACTIVITY, 2) SPIKING DUE TO THE THERMAL
40 REM    TRANSIENT, 3) FUEL ROD GAP RELEASE DUE TO CLAD PERFORATION, AND 4) ESCAPE
50 REM    OF FISSION PRODUCTS FROM THE FUEL PELLETS THROUGH DIFFUSION. THE SPIKING
60 REM    ACTIVITY IS ASSUMED TO BE A MULTIPLE OF THE NORMAL ACTIVITY. THE FUEL ROD
70 REM    GAP INVENTORY IS ASSUMED TO BE COMPLETELY RELEASED UPON CLAD PERFORATION.
80 REM    FUEL PELLET INVENTORY IS MODELLED BY A CONSTANT ESCAPE RATE MODEL.
90 REM    NOTE: ALL TIMES ARE IN UNITS OF HOURS.
100 REM
110 REM             INDEX NUMBER            ISOTOPE
120 REM                  1                   I-131
130 REM                  2                   I-132
140 REM                  3                   I-133
150 REM                  4                   I-134
160 REM                  5                   I-135
170 REM                  6                   KR-85M
180 REM                  7                   KR-85
190 REM                  8                   KR-87
200 REM                  9                   KR-88
210 REM                  10                  XE-131M
220 REM                  11                  XE-133
230 REM                  12                  XE-135
240 REM                  13                  XE-138
250 REM
260 REM----------------------------------------------------------------------------
270 REM
280 DIM AGAPO(13), APO(13), ACOOLO(13), DECAY(13), FACT(13)
290 DIM ACOOLTO(13), APTO(13), AFFTO(13), ACOOL(13),AFF(13), ASPIKE(13)
300 REM
310 REM     FRACTION OF FUEL ASSUMED FAILED
320 REM
330 Y=.01
340 REM
350 REM     CLEAN-UP RATE CONSTANT
360 REM
370 KP=.0792
380 REM
390 REM     FUEL PELLET ESCAPE RATE CONSTANT
400 REM
410 NU=.0000468
420 REM
430 REM     TIME AFTER TRANSIENT WHEN LET-DOWN(CLEAN-UP) IS RE-ESTABLISHED (HOURS)
440 REM
450 TO=1!
460 REM
470 REM     SPIKING RATIO (SPIKE ACTIVITY/NORMAL ACTIVITY)
480 REM
```

```
490 R=5
500 REM
510 REM     SIZE OF TIME STEP (HOURS)
520 REM
530 TSTEP=.25
540 REM
550 REM     VOLUME OF THE PRIMARY COOLANT SYSTEM (CC)
560 REM
570 VOL=3.426E+08
580 T=0!
590 REM
600 REM     END OF TRANSIENT ANALYSIS (HOURS)
610 REM
620 TEND=12!
630 REM
640 REM     INITIAL FUEL ROD GAP INVENTORIES (CURIES)
650 REM
660 AGAPO(1)=1.1E+07
670 AGAPO(2)=12000!
680 AGAPO(3)=1.1E+07
690 AGAPO(4)=2.4E+07
700 AGAPO(5)=1900000!
710 AGAPO(6)=2900000!
720 AGAPO(7)=91000!
730 AGAPO(8)=11!
740 AGAPO(9)=7500000!
750 AGAPO(10)=77000!
760 AGAPO(11)=2.1E+07
770 AGAPO(12)=3000000!
780 AGAPO(13)=1.8E+07
790 REM
800 REM     INITIAL FUEL PELLET INVENTORIES (CURIES)
810 REM
820 APO(1)=1.132E+08
830 APO(2)=1.654E+08
840 APO(3)=2.284E+08
850 APO(4)=2.467E+08
860 APO(5)=2.128E+08
870 APO(6)=2.854E+07
880 APO(7)=905500!
890 APO(8)=5.234E+07
900 APO(9)=7.478E+07
910 APO(10)=797100!
920 APO(11)=2.292E+08
930 APO(12)=4.102+7
940 APO(13)=1.828E+08
950 REM
```

```
960 REM    INITIAL PRIMARY COOLANT ACTIVITIES (MICRO-CURIES/MILLI-LITER)
970 REM
980 ACOOLO(1)=.198
990 ACOOLO(2)=.0732
1000 ACOOLO(3)=.278
1010 ACOOLO(4)=.0344
1020 ACOOLO(5)=.139
1030 ACOOLO(6)=.0805
1040 ACOOLO(7)=.11
1050 ACOOLO(8)=.0439
1060 ACOOLO(9)=.146
1070 ACOOLO(10)=.0805
1080 ACOOLO(11)=13.178
1090 ACOOLO(12)=.256
1100 ACOOLO(13)=.0322
1110 REM
1120 REM    DECAY CONSTANTS (INVERSE HOURS)
1130 REM
1140 DECAY(1)=.003591
1150 DECAY(2)=.303
1160 DECAY(3)=.0333
1170 DECAY(4)=.79
1180 DECAY(5)=.105
1190 DECAY(6)=.155
1200 DECAY(7)=7.369E-06
1210 DECAY(8)=.547
1220 DECAY(9)=.248
1230 DECAY(10)=.002409
1240 DECAY(11)=.00546
1250 DECAY(12)=.0756
1260 DECAY(13)=2.925
1270 REM
1280 REM    CONVERSION FACTORS (FROM MICRO-CURIES/MILLI-LITER TO COUNTS/MINUTE)
1290 REM
1300 FACT(1)=924400!
1310 FACT(2)=3017000!
1320 FACT(3)=877500!
1330 FACT(4)=2145000!
1340 FACT(5)=1247000!
1350 FACT(6)=775900!
1360 FACT(7)=398200!
1370 FACT(8)=1262000!
1380 FACT(9)=1169000!
1390 FACT(10)=18550!
1400 FACT(11)=322100!
1410 FACT(12)=860700!
1420 FACT(13)=2899000!
```

```
1430 REM
1440 REM    PRINT OUT THE INPUT DATA
1450 REM
1460 LPRINT CHR$(12)
1470 LPRINT "-------------------------------------------------------------------------------------"
1480 LPRINT " INDEX            INITIAL        INITIAL        INITIAL        CONVERSION     DECAY"
1490 LPRINT " NUMBER           PELLET         GAP            COOLANT        FACTOR         CONSTANT"
1500 LPRINT "                  INVENTORY      INVENTORY      INVENTORY"
1510 LPRINT "-------------------------------------------------------------------------------------"
1520 LPRINT " "
1530 FOR I=1 TO 13
1540 LPRINT I,APO(I),AGAPO(I),ACOOLO(I),FACT(I),DECAY(I)
1550 NEXT I
1560 LPRINT " "
1570 LPRINT " "
1580 LPRINT "PERCENT FAILED FUEL                                          ",Y*100!
1590 LPRINT "LETDOWN CLEAN-UP RATE COEFFICIENT                            ",KP
1600 LPRINT "DIFFUSION RATE COEFFICIENT OF FUEL PELLET                    ",NU
1610 LPRINT "TIME AFTER TRANSIENT WHEN CLEAN-UP BEGINS                    ",TO
1620 LPRINT "SPIKING FACTOR (SPIKE/NORMAL ACTIVITIES)                     ",R
1630 LPRINT "VOLUME OF PRIMARY SYSTEM                                     ",VOL
1640 LPRINT "TIME STEP SIZE                                               ",TSTEP
1650 LPRINT "END OF ANALYSIS TIME                                         ",TEND
1660 LPRINT CHR$(12)
1670 LPRINT "-----------------------------------------------------------------------------------------------------
---------------------------"
1680 LPRINT "TIME               STEADY         SPIKE          FAILED         TOTAL          STEADY         SPIKE
  FAILED         TOTAL"
1690 LPRINT "(HOURS)            STATE          ACTIVITY       FUEL           ACTIVITY       STATE          CPM
  FUEL           CPM"
1700 LPRINT "                   ACTIVITY                      ACTIVITY                      CPM
  CPM"
1710 LPRINT "-----------------------------------------------------------------------------------------------------
---------------------------"
1720 LPRINT " "
1730 REM
1740 REM    CALCULATE THE ACTIVITIES AT TIME TO
1750 REM
1760 FOR I=1 TO 13
1770 ACOOLTO(I)=ACOOLO(I)*EXP(-DECAY(I)*TO)
1780 APTO(I)=APO(I)*EXP(-TO*(DECAY(I)+Y*NU))
1790 AFFTO(I)=Y*AGAPO(I)*EXP(-DECAY(I)*TO)+APO(I)*(1!-EXP(-NU*TO))*EXP(-DECAY(I)*TO)*Y*NU/(DECAY(I)+NU)
1800 NEXT I
1810 REM
1820 REM    LOOP OVER TIME
1830 REM
1840 FOR J=1 TO 100
1850 IF T>TEND THEN GOTO 2390
```

```
1860 IF T>TO THEN GOTO 1990
1870 REM
1880 REM    FOR T<TO, CALCULATE THE TIME DEPENDENT ACTIVITIES
1890 REM
1900 FOR I=1 TO 13
1910 ACOOL(I)=ACOOLO(I)*EXP(-DECAY(I)*T)
1920 ASPIKE(I)=R*ACOOL(I)
1930 AFF(I)=Y*AGAPO(I)*EXP(-DECAY(I)*T)+APO(I)*(1!-EXP(-NU*T))*EXP(-DECAY(I)*T)*Y*NU/(DECAY(I)+NU)
1940 NEXT I
1950 REM
1960 REM    FOR T>TO, CALCULATE THE TIME DEPENDENT ACTIVITES
1970 REM
1980 GOTO 2070
1990 FOR I=1 TO 13
2000 ACOOL(I)=ACOOLTO(I)*EXP(-(DECAY(I)+KP)*(T-TO))
2010 ASPIKE(I)=R*ACOOL(I)
2020 AFF(I)=AFFTO(I)*EXP(-(DECAY(I)+KP)*(T-TO))+NU*Y*APO(I)*(EXP(-(DECAY(I)+KP)*(T-TO))-EXP(-(DECAY(I)+NU)*(T-
TO)))/(NU+DECAY(I))
2030 NEXT I
2040 REM
2050 REM    SUM ACTIVITIES AT TIME T
2060 REM
2070 TACOOL=0!
2080 TASPIKE=0!
2090 TAFF=0!
2100 FOR I=1 TO 13
2110 TACOOL=TACOOL+ACOOL(I)
2120 TASPIKE=TASPIKE+ASPIKE(I)
2130 TAFF=TAFF+AFF(I)
2140 NEXT I
2150 REM
2160 REM    SUM DETECTOR RESPONSES AT TIME T (CPM'S)
2170 REM
2180 TAFF=TAFF*1000000!/VOL
2190 TOTALA=TACOOL+TASPIKE+TAFF
2200 RCOOL=0!
2210 RSPIKE=0!
2220 RFF=0!
2230 FOR I=1 TO 13
2240 RCOOL=RCOOL+ACOOL(I)*FACT(I)
2250 RSPIKE=RSPIKE+ASPIKE(I)*FACT(I)
2260 RFF=RFF+AFF(I)*FACT(I)
2270 NEXT I
2280 RFF=RFF*1000000!/VOL
2290 RTOTAL=RCOOL+RSPIKE+RFF
2300 REM
2310 REM    PRINT ANSWERS
```

```
2320 REM
2330 LPRINT USING "##.###····     ";T,TACOOL,TASPIKE,TAFF,TOTALA,RCOOL,RSPIKE,RFF,RTOTAL
2340 REM
2350 REM    INCREMENT TIME
2360 REM
2370 T=T+TSTEP
2380 NEXT J
2390 LPRINT "END"
```

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