Philippine Engineering Journal (1992) XIII (2): 103-109

NUCLEAR ENGINEERING (NUC)

NUC 8101 **Rioflorido, Carmelo M. (MS Nuc. Eng'g.)**
A proposed inservice inspection program for the first Philippine Nuclear Power Plant (PNPP-1).
1981.

With the inception of the Philippines' commercial nuclear power industry, preparatory measures are needed to motivate and maintain public confidence in this extremely valuable energy source. One of these measures is the establishment of inservice inspection (ISI) program for the nuclear power plant throughout its 40-year lifetime. This thesis proposes an inservice inspection program for the country's first nuclear power plant (PNPP-1). The proposed ISI program has been developed based on the requirement of a widely adopted code for nuclear inservice inspection the section XI of the ASME Boiler and Pressure Vessel Code entitled "Rules for Inservice Inspection of Nuclear Power Plant Components".

The proposed ISI program complies with the requirements of the 1977 Edition of the ASME Section XI Code with addenda through the summer 1978. It provides the most acceptable licensee commitments, and the latest techniques currently being utilized abroad in the performance of an ISI. the proposed program is beneficial to the licensee in the sense that it could be adopted to the PNPP-1 with very slight modification. To the regulatory authority., the proposed ISI program is valuable in evaluating the adequacy of the ISI plans and schedules that will eventually be submitted by the licensee for approval.

NUC 8102 **Rodriguez, Perpetua I. (MS Nuc. Eng'g.)**
A study of the radioactive material release from the gaseous and liquid effluents of the first Philippine Nuclear Power Plant (PNPP-1).
1981.

Radioactive material releases from the gaseous and liquid effluents of the first Philippine Nuclear Power Plant (PNPP-1) were determined using a modified PWR-GALE Code. This code is a computer program employing a mathematical model for "source term" calculations for pressurized water reactors.

The input data utilized for source term calculations are based on the information gathered from the Preliminary Safety Analysis Report and the Environmental Report for the first Philippine Nuclear Power Plant. The computer output gives the annual release rates of the following: (1) noble gases, (2) radionuclides and radioactive materials in particulate form, (3) radionuclides other than noble gases, (4) corrosion and activation products, and (5) fission products. The results obtained from the computer output were used to make a comparative analysis of the annual release rates and radionuclide concentration with that of 10 CFR PART 20, to determine if the PNPP-1 has been adequately designed to reduce the radioactive releases to "as low as is reasonably achievable".

NUC 8203 **Puga, Gonzalo S. (MS Nuc. Eng'g.)**
A study of the patterns of maintenance and problems of nuclear medicine instrumentation in the Philippines.
1982.

The experimental investigation consists mainly of: (10 a survey of the nuclear medicine instrumentation, seeking informations on the structure of the medical unit, on instrument operating environment and maintenance, and on instrument operation reports; and (2) a monitoring if the voltage, temperature, and relative humidity conditions of the laboratories. The findings reveal significant factors which contribute to instrument inoperation and downtime, among which are: (1) not-so-ideal environment for optimum performance and long useful life of the instruments; (2) lack of protection for the instruments from the effects of AC mains fluctuations and disturbances; (3) undeveloped capability to institute/conduct repair and maintenance; (4) absence/incompleteness of appropriate service manuals; (5) unavailability of critical instrument components in the local market; and (6) some administrative constraints. Recommendations, based on the theoretical considerations previously discussed, are then formulated. At present, some degree of implementation of the recommendations have already been achieved.

In summary, the maintenance of nuclear instruments and of any other sensitive instruments, is not a simple endeavor, more especially in developing countries like the Philippines. Solutions to the various problems which effect instrument reliability could only be sought through the collaborative efforts of all concerned. To the maintenance technicians and engineers, this is, therefore, hoped that this thesis could help, even in just a small way, in providing a systematic approach towards an increased reliability of instruments.

NUC 8304 **Noriel, Marilou Claire J. (MS Nuc. Eng'g.)**
A study of annual doses to man from routine gaseous effluents releases of the Philippine Nuclear Power Plant Unit I (PNPP-1)
1983.

Individual and population integrated doses from radioactive gaseous releases of PNPP-1 were calculated using a modified GASFAR Code. Input data consisted of meteorological and site data gathered from the PNPP-1 Final Safety Analysis Report (FSAR), population and agricultural production data from the National Economic and Development Authority (NEDA) and the National Census and Statistics Office (NCSO). Usage factors were calculated based on Food and Nutrition Research Institute (FNRI) recommended dietary allowances for Filipinos. Results of population integrated dose calculations were used in identifying the critical nuclides, the critical body organs, and the critical pathway. Results from individual dose calculation were used in determining compliance with the dose limits set forth in Appendix D of Part 7 Code of PAEC Regulations.

NUC 8405 **Villanueva, Eliseo Piodos (MS Nuc. Eng'g.)**
Fault tree analysis of the Emergency Core Cooling System of the Philippine Nuclear Power Plant Unit No. 1.
1984.

Postulated undesired state of the Emergency Core Cooling System (ECCS) of the first Philippine Nuclear Power Plant Unit No. 1 (PNPP-1) was specified following a loss of coolant accident (LOCA). The ECCS was analyzed and a fault tree was generated. A computer was developed to process the fault tree.

The input data utilized in the study are based on the information gathered from the Safety Analysis Report for the PNPP-1 and the USNRC Reactor Safety (WASH-1400). The computer outputs are event probabilities and a listing of the minimal cut sets causing failure arranged by single failures, then double, triple, etc. The results obtained from the computer outputs determine the combination of component failures that would cause the occurrence of the top event of the fault tree and the spots in the subsystems of the ECCS.

NUC 8506 **Generoso, Reynaldo Ty. (MS Nuc. Eng'g.)**
A computing method to predict the conditions at which a nuclear reactor core attains criticality.
1985.

It is the intention of very electric utility system to maximize the availability of its power plants. This concern is important specifically in nuclear power plants where operations are complex and energy costs are considerable. One of several ways of effecting this is by minimizing shutdown time through a quick return to power after a trip. The study at hand purposes a calculational tool that can facilitate planning for efficient startup recovery of a nuclear power plant.

The study, which resembles of a computer software development project, is organized into five phases: (1) requirements analysis, (2) preliminary design, (3) detailed design, (4) coding, and (5) testing. The accomplishment in each phase are described in separate chapters of this research. They are the (1) software requirements specification, (2) functional design specification, (3) detailed design document, (4) computer program listing, and (5) test document.

The software requirements specification presents an overview of the theoretical basis for estimating the conditions at which a nuclear reactor core attains criticality. A manual procedure used by reactor operators is described and evaluated. Possible areas for improvement are identified and the requirements for a new computing method are outlined. A set of criteria is defined to demonstrate the correctness of the resulting program.

The functional design specification presents the functions that the program performs starting with the most general function going down to the most detailed functions. The equations and algorithms that result from this specification constitute the bases for subsequent detailed design.

The detailed design document presents a complete design and operational description of the program. Input-processing-output charts are used extensively to describe to process in sufficient detail for program coding. The organization of the external data bank is also presented.

The computer program listing presents the program a source code level of detail.

The test document serves to formally demonstrate that the program satisfies the software requirements and design specifications. Test activities are performed at three levels: (1) program unit testing, (2) module integration testing, and (3) validation testing. These tests show that results from the program are in good agreement with the best estimates from the manual procedure. Furthermore, the calculation time is reduced sixtyfold.

The proposed calculational tool is called the ECCMAIN program.

NUC 8507 **Pinlac, Sabino M. (MS Nuc. Eng'g.)**
A study of the PNPP-1 Spent Fuel Pool Heat Balance using a Computer-Aided Simplified Calculational
1985.

This study presents a simplified means of evaluating the capabilities of the PNPP-1 Spent Fuel Pit Cooling and Makeup System to perform their functions. The evaluation is based on data generated by running five computer programs on the TRS microcomputer system. These data, generally fall into five groups, four of which were used to evaluate the effectiveness of the spent fuel cooling process. These are the spent fuel residual decay heat generation rates, spent fuel pool average temperatures, natural circulation fuel channel coolant mass flow rates, and the maximum fuel cladding temperatures of the hottest spent fuel batch. The fifth group of data are the spent fuel pool evaporation rates which served as the basis from which the spent fuel pit makeup water system capability was evaluated.

The spent fuel residual decay heat generation rates were calculated based on a continuous reactor rated power operation of one year in between refueling and for cooling times starting at 150 hours after the last reactor shutdown for refueling. This data group is further classified into two; a) decay heat generation rates of all the spent fuel batches stored in the spent fuel poll, and b) decay heat generation rates of the hottest spent fuel batch. The former was used to determine the amount of heat that has to be dissipated in the heat exchangers, and also for the calculation of instantaneous pool average temperatures. The latter was used in the determination of the fuel channel coolant mass flow rates and accordingly, the maximum fuel clad temperatures.

The spent fuel pool average temperatures were calculated for two different heat loadings; a) heat load from the spent fuel batches, and b) heat load from thirteen spent fuel batches. In each of these heat loadings, three different operating conditions were evaluated namely; a) complete loss of the spent fuel cooling and makeup system capability, b) one cooling train operating together with the makeup water system, and c) fully operable spent fuel cooling and makeup water system.

The natural circulation coolant mass flow rates along the hottest fuel channel were determined for pool average temperatures ranging from 120½F to 212½F, and for various cooling times after the last reactor shutdown.

The maximum fuel cladding temperatures of the hottest spent fuel element were calculated based on the range of calculated coolant mass flow rates and for pool average temperatures ranging from 120½F to 212½F.

Pool evaporation rates were calculated for poll average temperatures ranging from 120½F to 212½F, and for relative humidity values ranging from 0.60 to 0.90.

The results of the calculations showed that:

a. When thirteen spent fuel batches are placed in the spent fuel pool, two cooling trains should be operated in order to maintain the pool temperature at or below 150½F.

b. When ten spent fuel batches are placed in the spent fuel pool, the operation of one cooling train in sufficient to maintain the pool temperature at or below 120½F.

c. The spent fuel cladding temperatures attained for the different conditions considered are less significant as compared to the fuel cladding temperatures attained in a reactor running at full power.

d. The makeup water requirement for the extreme case is about 300 gallons per hour. This condition is characterized by a pool average temperature of 212½F and a relative humidity value of 0.60.

e. The most serious case that may occur is not the boiling of the spent fuel pool but its drying up since this will surely result to the liberation of the fission gases to the spent fuel building and ultimately to the atmosphere.