MAY 14 1965

GEAP-4842
APRIL 1, 1965

EURAEC
JOINT U.S.-EURATOM RESEARCH
AND DEVELOPMENT PROGRAM

1373

MASTER

DEVELOPMENT PROGRAM ON THE GARIGLIANO NUCLEAR REACTOR

QUARTERLY REPORT NO. 10

C.L. HOWARD

PERMIT CLEARANCE OBTAINED. RELEASED
THE PUBLIC IS APPROVED. PROCEDURES
ARE ON FILE IN THE RECEIVING SECTION.

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189
PROJECT AGREEMENT 33

ATOMIC POWER EQUIPMENT DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

DEVELOPMENT PROGRAM ON THE
GARIGLIANO NUCLEAR REACTOR

QUARTERLY REPORT NO. 10

C. L. Howard

Prepared Under
U.S. ATOMIC ENERGY COMMISSION
Contract AT(04-3)-189
Project Agreement 33

for the
JOINT U.S.-EURATOM
Research and Development Program

*Printed in U.S.A. Available from the
Clearing House for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia
Price: \$2.00 per copy*

ATOMIC POWER EQUIPMENT DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

LEGAL NOTICE

THIS DOCUMENT WAS PREPARED UNDER THE SPONSORSHIP OF THE ATOMIC ENERGY COMMISSION PURSUANT TO THE JOINT RESEARCH AND DEVELOPMENT PROGRAM ESTABLISHED BY THE AGREEMENT FOR COOPERATION SIGNED NOVEMBER 8, 1958, BETWEEN THE GOVERNMENT OF THE UNITED STATES OF AMERICA AND THE EUROPEAN ATOMIC ENERGY COMMUNITY (EURATOM). NEITHER THE UNITED STATES, THE U.S. ATOMIC ENERGY COMMISSION, THE EUROPEAN ATOMIC ENERGY COMMUNITY, THE EURATOM COMMISSION, NOR ANY PERSON ACTING ON BEHALF OF EITHER COMMISSION:

- A. MAKES ANY WARRANTY OR REPRESENTATION, EXPRESS OR IMPLIED, WITH RESPECT TO THE ACCURACY, COMPLETENESS, OR USEFULNESS OF THE INFORMATION CONTAINED IN THIS DOCUMENT, OR THAT THE USE OF ANY INFORMATION, APPARATUS, METHOD, OR PROCESS DISCLOSED IN THIS DOCUMENT MAY NOT INFRINGE PRIVATELY OWNED RIGHTS; OR
- B. ASSUMES ANY LIABILITIES WITH RESPECT TO THE USE OF, OR FOR DAMAGES RESULTING FROM THE USE OF ANY INFORMATION, APPARATUS, METHOD OR PROCESS DISCLOSED IN THIS DOCUMENT.

AS USED IN THE ABOVE, "PERSON ACTING ON BEHALF OF EITHER COMMISSION" INCLUDES ANY EMPLOYEE OR CONTRACTOR OF EITHER COMMISSION OR EMPLOYEE OF SUCH CONTRACTOR TO THE EXTENT THAT SUCH EMPLOYEE OR CONTRACTOR OR EMPLOYEE OF SUCH CONTRACTOR PREPARES, HANDLES, DISSEMINATES, OR PROVIDES ACCESS TO, ANY INFORMATION PURSUANT TO HIS EMPLOYMENT OR CONTRACT WITH EITHER COMMISSION OR HIS EMPLOYMENT WITH SUCH CONTRACTOR.

To expedite dissemination of technical information to those persons requiring it, this report has been issued without the administrative review necessary to protect the interest of the U.S. Government, the European Atomic Energy Community (Euratom) and the contractor. The recipient of this report agrees that it is passed on to him in confidence, that he will maintain that confidence and make no disclosures or publication of the contents, that would be detrimental to the interests of the U.S. Government, the European Atomic Energy Community (Euratom) and the contractor.

TABLE OF CONTENTS

	<u>Page</u>
INTRODUCTION	-1-
SUMMARY	-2-
PROGRAM ASSIGNMENTS	-3-
STATEMENT OF PROBLEM	-4-
TASK I - DATA LOGGING AND COMPUTER SYSTEM	-5-
TASK II - PRESSURE VESSEL SAMPLE IRRADIATIONS	-12-
TASK III - INSTRUMENTED ASSEMBLIES	-13-
TASK IV - SPECIAL STUDIES AND TESTS	-15-
A-1, Core Evaluation Studies	-15-
A-2, Improved Measurement of Gross Power Distribution	-15-
A-3, Safeguards Studies	-16-
B-1, Stability Tests and Analysis	-19-
B-2, Plant System Tests and Analysis	-20-
TASK V - COORDINATION	-24-
PREVIOUS REPORTS	-25-
REFERENCES	-27-

INTRODUCTION

The United States and the European Atomic Energy Community (Euratom), on May 29, and June 18, 1958, signed an agreement which provides a basis for cooperation in programs for the advancement of the peaceful applications of atomic energy. This agreement, in part, provides for the establishment of a Joint U.S.-Euratom research and development program which is aimed at reactors to be constructed in Europe under the Joint Program.

The work described in this report represents the Joint U.S.-Euratom effort which is in keeping with the spirit of cooperation in contributing to the common good by the sharing of scientific and technical information and minimizing the duplication of effort by the limited pool of technical talent available in western Europe and the United States.

SUMMARY

1. Programming for the on-line computer has been completed, and the programs are being tested at the reactor site. Computer installation also has been completed.
2. The program for on-line calculation of the neutron flux distribution has been tested, using better albedo values, and was found to give good results. A minor change in the procedure for usage of the 80 in-core ion chambers gives further improvement.
3. The instrumentation for the four instrumented assemblies is being fabricated. Use of movable ion chambers adjacent to the corners of the instrumented fuel assemblies has been approved by SENN, and detail design is in progress.
4. A different method of measuring power distribution in irradiated fuel has been proposed, and equipment is being designed. The method uses the $D_2O (\gamma, n)$ reaction and measurement of the resulting neutrons to obtain the profile of La-140 activity in the fuel.
5. Additional safeguards data have been submitted to SENN for use in obtaining CNEN approval for the reactor tests.
6. The control rod oscillator equipment has been assembled, and is being tested in San Jose. The stability pre-test analysis has been slightly revised to use the best estimates of recirculation flow and inlet subcooling.
7. SENN is in the process of being nationalized. The responsibilities of SENN in the Development Program will be assumed by ENEL.

PROGRAM ASSIGNMENTS

Project Engineer (Task V, Coordination)
Task I, Data Logging and Computer System

A. System Specifications

B. Programming

Task II, Pressure Vessel Sample Irradiation

Task III, Instrumented Assemblies

Task IV, A-1, Core Evaluation Studies

Thermal-Hydraulics

Physics Analysis

A-2, Gross Power Distribution

A-3, Safeguards Studies

B-1, Stability Tests and Analysis

B-2, Plant System Tests and Analysis

C. L. Howard

L. K. Holland

{ T. Sorlie
N. H. Barth
R. T. Bellan

{ S. L. Gummersall
H. S. Sakasegawa
T. Mazzanti*

I. R. Kobsa

F. A. Brandt

F. L. Abboud

A. C. Duckart
S. Viscuso*

R. K. Baird

{ W. V. Macnabb
H. E. Townsend

{ D. L. Fischer
N. H. Barth
G. duBoucheron**

D. L. Fischer

N. H. Barth
W. R. Morgan

C. L. Howard

F. Santasilia*
S. Viscuso*
A. Colling**
L. H. Youngborg
T. Sorlie
W. I. Collett
A. C. Duckart
L. K. Holland
N. H. Barth
R. T. Bellan

L. K. Holland

L. H. Youngborg
S. Viscuso*
A. Colling**

*SENN engineer working at San Jose, California
**Euratom participant

STATEMENT OF PROBLEM

The development program on the Garigliano Nuclear Reactor is directed toward the following objectives: (1) to accumulate data, above that usually obtained by the utility, for safe, efficient, and reliable plant operation; (2) to evaluate such data; and (3) to investigate the feasibility of improved plant performance. The specific aspects of plant performance to be investigated are indicated by the titles of the tasks.

TASK I - DATA LOGGING AND COMPUTER SYSTEM

This task deals with the specification of a data logging and computer system for installation at the Garigliano plant. The system will detect and periodically print preselected variables such as signals from in-core ion chambers; condenser, feedwater, and generator data; and exposure of fuel bundles and control rods. The data logging and computer system will also reduce the measured data, calculate critical variables, and alarm all those variables that exceed prescribed limits.

Specification of the computer system and calculational logic and equations have been performed by GE-APED as part of the Development Program. The computer has been purchased by SENN, as part of the Development Program, from International General Electric. APED and SENN are jointly conducting the computer programming.

The computer installation and computer acceptance test were essentially completed in March 1965.

Program testing and debugging have been in progress at the reactor site since February 23. Prior to that time, the programming of all calculations and logs was completed in San Jose.

The following calculational modules have been tested at the reactor site and are now considered operational, except for possible minor errors:

- Monitor,
- Plant performance calculations,
- On-demand logs,
- Hourly log, and
- Scan programs -- partially tested.

Program work at the Site will continue for another two or three months.

The power distribution calculation for the on-line computer at SENN has been compared with observed reactor flux shapes, and with diffusion theory calculations. The feature that allows in-core flux monitor value domination of the power shape shows excellent promise as a valuable reactor power shape determination.

The calculation, as performed on the SENN on-line computer, is a few-node, one-group diffusion theory calculation modified for reflector effects in both the radial and axial directions. These effects are best determined from observed data, wire irradiations, or gamma scans, and thus far

have not yielded to derivation. The radial effects have been obtained using the data of the last gamma scan, and are yielding satisfactory results in observable effects.

A high sensitivity to axial reflector effects has been noticed. This produces marked axial shape variations with small changes of input, or with changes of over-all reactor void content, such as in conditions of low flow or power uprating.

A series of calculations was performed, in a one-dimensional version of the SENN program, of a typical high void channel, demonstrating the effect of vertical albedo changes over a small (in an arithmetical sense) range. The differences are shown in Figure 1, in which axial node powers (a part of a series) are represented as lines and the only variables are the axial reflector effects which are the same at top and bottom. Shown also is the standard, three-group diffusion theory calculation of the same problem, with the water reflector taken into account. From these calculations, and observations of three-dimensional cases, it is possible to choose the reflector effects that will yield the best agreement over the widest range of operating conditions. This best agreement appears to be, for SENN, approximately zero correction.*

The second set of calculations presented demonstrates the apparent power of the technique which allows in-core flux monitor values to be used as known points of information. Monitor flux is corrected for adjacent control rod insertion to yield effective adjacent bundle power averages. A program modification has been made to use the monitor reading as the "correct value" for the average of the two adjacent axial nodes instead of one as specified in GEAP-4356, "Physics of the On-Line System."

As a trial, the reactor configuration extant June 1, 1964, was computed. The results for four specific irradiated wires are presented in Figures 2 through 5, which show the comparison of wire axial irradiation shape, the SENN on-line unaltered version for the same core location, and the on-line calculation result when the in-core monitor values (from the wire data) were included. The average power of the four surrounding bundles is considered to be what the monitor indicates, and is presented as the comparison. The figures show good agreement between the wires and the computer result, considering the difficulty in representing a two-hump curve by eight nodes. Further correction of in-core monitor values for additional effects which would affect bundle power-to-flux ratios are easily developed if experience indicates they are necessary.

*The reflector effect, or albedo, is incorporated in the calculations by (1) using zero as the value of the source for the fictitious node outside the core, and (2) multiplying the calculated flux for nodes 1 and 8 by a reflector correction term (for example, $1.06 = 1 + \alpha_K \beta$) before storing the value for the next iteration.

Figure 1. Effect of Albedo on Axial Power Shape

Figure 2. Comparison of Measured and Calculated Axial Flux Profile - Wire Location 101

Figure 3. Comparison of Measured and Calculated Axial Flux Profile - Wire Location 106

Figure 4. Comparison of Measured and Calculated Axial Flux Profile - Wire Location 113

Figure 5. Comparison of Measured and Calculated Axial Flux Profile - Wire Location 115

TASK II - PRESSURE VESSEL SAMPLE IRRADIATIONS

This task is concerned with the irradiation behavior of reactor pressure vessel materials. It consisted of the preparation of about 870 pressure vessel samples. The samples were made from a "standard heat material" and from actual material used to fabricate the Garigliano reactor vessel, including tangential and longitudinal base metal, weld metal, and heat-affected zone base metal. A total of about 765 Charpy and tensile specimens were encapsulated and installed inside the steam drum, between the reactor core and thermal shield, and between the thermal shield and the pressure vessel wall.

This General Electric Company portion of this task has been completed except for consultation on the interpretation of results. The first samples will be removed from the reactor at the next refueling shutdown, about July 1965, and a Euratom laboratory will conduct tests on the samples.

TASK III - INSTRUMENTED ASSEMBLIES

This task provides for the design, fabrication, calibration, supervision of installation, and evaluation of four fuel assemblies instrumented to measure channel flow rates, water inlet and outlet temperatures, and neutron flux.

Detailed drawings of the flow and temperature portions of the Instrumented Fuel System are nearing completion. The status of hardware and instruments is as follows:

Flowmeters	On order*
Thermocouples	On hand
Probe	Drawings released to Manufacturing
Cable clamps	Check prints being reviewed
Vessel flange	Check prints being reviewed
Thermocouple reference junction	On hand
Readout panel	
Assembly drawings	Drawings released for manufacture
Cabinet	On hand
Low voltage power supplies	Released for manufacture
Flow signal conditioners	Manufactured
Flow ratemeters	Released for manufacture
Thermocouple amplifiers (MV/I)	On order
Temperature indicators	Released for manufacture
In-core flux converters	Released for manufacture
Recorders	On order*

Use of four SENN spare fuel bundles for the instrumented fuel assemblies has been proposed as an alternate to the original intention of manufacturing four new bundles as part of the Development Program. The terms and conditions of such use of SENN fuel are under negotiation with SENN and the AEC.

The design of the four movable in-core flux chambers (see Quarterly Report No. 8) has been modified to include in each unit:

- a. Four fixed ion chambers,
- b. One wire irradiation guide tube, and
- c. One movable ion chamber.

*Temporary AEC "hold" on these purchase orders.

This overcomes the previously-discussed problem of the method of calibrating the fixed in-core chambers in case the movable chamber failed. Now the calibration of the fixed chambers is accomplished by use of the normal wire irradiations, and the operability (or lack of operability) of the movable chambers is not a problem.

The status of the design and procurement is as follows:

- a. Subminiature ion chamber - Released for parts procurement,
- b. Fixed in-core chambers and standard hardware - Released for parts procurement,
- c. Movable chamber drive - Being designed,
- d. Drive control and position display module - Released for parts procurement.

The movable chamber is a 0.125-inch o.d. by 0.247-inch long, boron-coated, neutron-sensitive ion chamber. It is attached to a 35-foot-long, 0.125-inch o.d., stainless-steel-sheathed, alumina-insulated cable.

The use of a boron coating on the chamber ensures that the maximum operating temperature (650°F) will not be exceeded. Depletion of the boron with accumulated neutron exposure will reduce the chamber sensitivity to 50 percent of initial sensitivity after about 1.9×10^{20} nvt exposure. This means that if the chamber is left in the peak reactor flux for about 70 hours, the sensitivity of the chamber will be reduced by about 5 percent.

Sensitivity versus exposure follows the equation:

$$S = S_0 e^{\left(\frac{-nvt}{2.7 \times 10^{20}} \right)},$$

where

S = sensitivity at time t,
 S_0 = sensitivity at time 0,
 nv = neutron flux (thermal),
 t = time.

Because of the decrease in sensitivity with exposure, it is recommended that the chambers be retracted out of the high-flux region when measurements are not being made.

Calibration of the movable chambers may be made by following the same procedures used for wire irradiations; the integral flux curve is equated to channel power to normalize the values. In this case the channel power level may be determined from the measured values in the adjacent instrumented channel, where inlet subcooling, Δh_s , is determined from the thermocouples, and flow and exit quality are determined from the flow measurements. Flux wire irradiations also can be made in the new flux detector assemblies, as in the standard assemblies. Traverses with the movable chambers may be made during wire irradiations and the results compared.

TASK IV - SPECIAL STUDIES AND TESTS

A-1, Core Evaluation Studies

This task provides for evaluation of the physics, heat transfer, and hydraulics aspects of various methods to attain high power density. After limiting problem areas, economic incentives, and potential for developmental success of the various concepts have been evaluated, an approach well suited to the Garigliano core will be proposed. Final evaluation and implementation of any plant uprating are, of course, made by SENN.

A topical report on the Thermal-Hydraulic Analysis of the Garigliano core has been written, and is now in final typing. The physics analysis has been delayed until completion of the testing of the on-line computer methods, as described under Task I. The same computer code will be used for the physics portion of the core evaluation.

A-2, Improved Measurement of Gross Power Distribution

This task provides for use of ion chamber, instrumented assembly, and gamma probe data (obtained by SENN) to formulate representations of the gross power distributions at various operating conditions. The results will be compared to predicted values obtained with the on-line computer, and also the results will provide input to the core evaluation study.

The first measurements will be made in the summer of 1965, and will be repeated at shutdowns in 1966 and 1967. Several methods of making the measurements have been investigated.

The measurement of the power distribution of a reactor after shutdown has been accomplished by the use of one of several techniques. A technique that is suitable for use in the field consists of an ionization chamber reading of the gamma dose with respect to position. This technique has been described as a gamma probe. Interpretation of the data thus obtained is difficult as the dose rate and the chamber response depend on gamma energy of the source, and the source consists of fission products in varying stages of saturation and decay.

An improved technique is to observe the La^{140} activity. La^{140} is a member of mass 140 fission product chain and is in transient equilibrium with its parent, 12.8 day Ba^{140} . La^{140} decays by beta emission, followed by gammas of 1.6 MeV and 2.55 MeV, and others. A few days after

reactor shutdown, these gammas are more energetic than most of the remaining fission product gammas and can be separated by gamma spectroscopy or by threshold reactions.^(1, 2) The 12.8-day half-life allows interpretation in terms of the reactor power during the last 30 days or so of operation, and gives the experimenter time to perform the measurements.

Work is progressing to allow measurements of the gamma probe type and the type reported by M. L. Batch, et al.⁽²⁾ The current status of this work includes design of detector chambers, a gamma transducer and the mechanism to traverse the fuel element with the detector system, and placement of an order for fast electronics. The gamma transducer consists of a $2 \times 6 \times 12$ inch volume of D_2O . A gamma ray of energy greater than 2.2 MeV can react with deuterium and produce a neutron. The neutrons produced by this reaction are thermalized in the D_2O and surrounding water. Fission chambers are located on two sides of the D_2O chamber to detect these neutrons. The output of these chambers is then proportional to the La^{140} fission product distribution and, hence, to the recent power distribution in the fuel element. Fission chambers were selected because of their inherent gamma rejection. This, coupled with the fast collection time of these chambers (40 nsec) and fast electronics to avoid pulse pile-up, will allow operation in the high (about 10^6 R/h) gamma field.

The detector assembly will be enclosed in a cadmium-covered lead shield to prevent the fission chambers from detecting neutrons from the fuel and surrounding water, and to provide position resolution of approximately 1 inch. The scanning will be performed in the reactor fuel handling pool by constructing a fuel bundle support and a mechanism to traverse the detector adjacent to the fuel bundle. A preliminary sketch of the device is shown in Figure 6.

A-3, Safeguards Studies

The Safeguards task provides for analysis of the various accidents associated with the conduct of the development work, and the evaluation of the consequences of these accidents. The information is used by SENN to obtain the necessary approvals (CNEN) for performance of the tests.

Additional safeguards information was submitted to SENN in the form of a report, "Safeguards Report - Garigliano Development Program," compiled by C. L. Howard, March 15, 1965. The summary of this report is as follows:

In a previous report, "Preliminary Safeguards Report - Garigliano Development Program," November 30, 1964, the general plan of proposed tests on the Garigliano reactor was presented. Temporary changes in the Plant Technical Specifications were requested to permit performance of the tests at the desired reactor conditions.

This report provides additional data on the proposed tests, the test equipment, possible malfunctions of the test equipment, test procedures and safety criteria, calculational uncertainties, effects of "normal" plant transients, and results of special safety analyses which have been conducted. The more important conclusions are as follows:

Figure 6. Diagram of D_2O (γ, n) Method for Fuel Gamma Scan

1. The control rod oscillator and its control and safety circuit have been designed to be fail-safe with regard to both equipment malfunction and operator error. Analysis of possible accidents associated with the system shows that the protective devices are adequate to prevent damage to the reactor system and fuel, even if these accidents occur.
2. The four instrumented assemblies have no direct safety problems, except stoppage of the turbine flowmeters. Both the sizing of the inlet orifice to set the channel flow rate, and reactor calculational procedures will assume that the turbines have already stopped. Thus, no hazard exists even if both inlet and outlet turbines stop at the same time.
3. A previously discussed problem with the movable in-core flux chambers has been overcome by placing four fixed in-core chambers, one movable chamber and one flux wire irradiation tube in the same assembly. Thus, the possible failure of the movable chamber mechanism causes no change in normal plant procedures for calibrating and using the fixed in-core chambers.
4. A list of procedural safety criteria has been developed to ensure that these items are incorporated in the final detailed procedures.
5. Analysis of normal plant transients, such as generator breaker trip, at the most severe of the proposed test conditions shows that the peak of the fuel heat flux transient is only 10% which is well within the 20% overpower calculational allowance.
6. An extensive analysis of experimental error in determination of key reactor parameters has been conducted (assuming recent calibration of plant instruments and use of the most accurate plant instruments available) with the following results:

<u>Variable</u>	<u>Probable Error (0.675σ) %</u>	<u>Maximum Probable Error (1.645σ) %</u>
Reactor Total Power	0.62	1.51
Recirculation Flow (Core Total Flow)	0.94	2.29
Channel Flow Rate	1.15	2.80
Local Fuel Heat Flux	4.64	11.3
Minimum Critical Heat Flux Ratio	7.76	18.9

7. The minimum critical heat flux ratio (MCHFR) is then as follows at the limiting reactor condition of one loop natural circulation at low subcooling, and at rated power of 506 MWt:

	<u>At Actual Power</u>	<u>At 120% of Actual Power</u>
Expected Value of MCHFR	1.93	1.50
Maximum Probable Error	<u>-0.31</u>	<u>-0.31</u>
Lowest Probable Value, After Allowing for Maximum Probable Error	1.62	1.19

Thus, at the actual reactor power the MCHFR is above 1.62 and if an overpower transient of 20% occurs (and this is very rare) the MCHFR is above 1.19.

8. The precautions taken against problems of reactor stability are adequate. First, pre-test calculations have been made which show that the reactor is stable at the proposed test conditions. Second, the experimental procedures and high flux scram settings are adequate to provide protection even if the reactor were unstable.
9. The flow and heat transfer for the water outside the fuel channels have been analyzed. The amount of boiling and void formation has been determined and physics calculations of void coefficient and control rod worth indicate that no safety problems will be encountered.
10. The reactor has already been operated at several low flow conditions. The comparison of previous and proposed conditions is as follows:

	<u>Previous Operation</u>	<u>Proposed Tests</u>
Highest Power at Rated Flow, MWt	506	506
Highest Power at 2 Loops Natural Circulation, MWt	388	506
Highest Power at 1 Loop Natural Circulation, MWt	306	506*

*Or highest permitted by MCHFR limit.

11. The pre-test calculations presented here are for the purpose of establishing the credibility of conducting the proposed tests. The safety of the reactor at each test condition is to be justified at the time of the tests by calculations based on measured values of flow, power and flux distribution.
12. After conducting these analyses of the proposed tests, it is concluded that:
 - (a) The proposed temporary changes in the plant Technical Specifications are safe, and
 - (b) The tests themselves, including both equipment and procedural safeguards are safe with regard to both the reactor system and the fuel.

This report is to be discussed with SENN and CNEN early in April.

B-1, Stability Tests and Analysis

Steady-state and control rod oscillator tests are to be planned and conducted at various reactor operating conditions. Data evaluations will provide information on the stability of operation at increased power density and increased void content.

The rod oscillator equipment, safety criteria, and accident analysis have been presented in the Safeguards Report. Also, the equipment description and the calculated results of a plant stability analysis were given in Quarterly Report No. 9.

Since the previous analysis, improved calculations have been made. These calculations indicate that

- a. The reactor natural-circulation flow is slightly higher than calculated before — 7.25×10^6 lb/h instead of 6×10^6 lb/h.
- b. The subcooling is less than shown previously. This is due to decreased secondary steam generator load when one loop is out of service and at reduced recirculation flow rates. Also, it may not be possible to bypass all four feedwater heaters at the same time, thus further reducing the subcooling.

Those revised flow and subcooling values have been used in a new stability analysis. The results are shown in Table 1 and Figure 7. The peak response is slightly less than the previously calculated results.

Preliminary tests of the rod oscillator system are being performed at the control rod drive test facility at San Jose, California. Typical results of the tests are shown in Figure 8.

An amplitude of slightly less than 6 inches peak-to-peak may be obtained at frequencies up to 1 cycle per second. The amplitude limits for each frequency will be determined in the remaining tests.

B-2, Plant System Tests and Analysis

Transient and load-following tests are to be planned and conducted at various plant operating conditions. These tests will provide data on reactor operation at conditions simulating an amount of plant up-rating indicated by the core evaluation study. The tests are scheduled for the summer of 1965 and will be coordinated with the stability tests.

The following preliminary procedures for the tests have been prepared:

- a. Pressure transient "step change", and
- b. Pump trip test.

TABLE I

GARIGLIANO REACTOR AT FULL POWER,
ONE-LOOP NATURAL CIRCULATION

Subcooling	Subcooling (Btu/lb)	Feedwater and Secondary Condition			Reactor Inlet Temp. (°F)	Power (MW(t))	Recir- culation Flow (lb/h)	Gain and Phase Margin			
		Feedwater Heater	Secondary Power	Secondary Steam Flow (lb/h)				Gain	At cps	Phase	At cps
Low	61	All 4 on	0	0	496	512	7.27×10^6	13.0	0.614	79.0°	0.260
Rated	86	All 4 on	68.75 MW(t)	0.276×10^6	475	511	7.22×10^6	10.3	0.618	55.6°	0.331
High	122	All 4 off	68.75 MW(t)	0.204×10^6	443	504	7.36×10^6	6.9	0.634	34.6°	0.422

Figure 7. Garigliano Calculated Transfer Function (Flux Response,
 $\% / \text{Reactivity } \frac{\Delta k/k}{\beta}$, cents)

Figure 8. Control Rod Oscillator Data - 1 cps

TASK V - COORDINATION

1. A meeting in Rome, Italy, with SENN, Euratom, and AEC representatives, was held on January 18-20, 1965. The major subjects were: Safeguards criteria for the reactor tests, and proposed use of four SENN fuel bundles for the instrumented assemblies. Personnel attending were S. Levy, D. A. Eldred, and C. L. Howard of GE, and D. L. Layton of IGEOA.
2. Mr. Armand Colling arrived in San Jose on January 4 for a 4- to 6-month assignment as a Euratom participant. He is working on reactor test preparations and safeguards.
3. Mr. Gerard du Boucheron completed his 2-month assignment as a Euratom participant on January 22, 1965. He worked primarily on the physics portion of the SENN on-line computer code and, in particular, on the evaluation of the effect of albedo on power shape.
4. Computer programmers Messrs. T. Mazzanti (SENN) and S. Gummersall left San Jose February 11 and February 19, respectively, to begin computer program testing at Garigliano.
5. Mr. Franco Santasilia (SENN) arrived from Italy February 23, and stayed until March 20, working primarily on the Safeguards Report.
6. SENN is in the process of being nationalized by the Italian government. SENN will be absorbed into ENEL (Ente Nazionale per L'Energia Elettrica) which is the governmental body having responsibility for all electrical power generation and distribution in Italy. All SENN responsibilities under the Development Program will be assumed by ENEL, and in most cases the personnel doing the work will be the same as before.

PREVIOUS REPORTS

Quarterly Reports

1. GEAP-4164/EURAEC-528, "Development Program for Increased Output in the Garigliano Nuclear Reactor, Quarterly Report No. 1, " September-December 1962.
2. GEAP-4218/EURAEC-643, "Development Program for Increased Output in the Garigliano Nuclear Reactor, Quarterly Report No. 2, " January-March 1963.
3. GEAP-4306/EURAEC-752, "Development Program for Increased Output in the Garigliano Nuclear Reactor, Quarterly Report No. 3, " April-June 1963.
4. GEAP-4389/EURAEC-839, "Development Program for the Garigliano Nuclear Reactor, Quarterly Report No. 4, " July-September 1963.
5. GEAP-4486/EURAEC-941, "Development Program for the Garigliano Nuclear Reactor, Quarterly Report No. 5, " October-December 1963.
6. GEAP-4607/EURAEC-1028, "Development Program on the Garigliano Nuclear Reactor, Quarterly Report No. 6, " January-March 1964.
7. GEAP-4645/EURAEC-1134, "Development Program on the Garigliano Nuclear Reactor, Quarterly Report No. 7, " April-June 1964.
8. GEAP-4726/EURAEC-1246, "Development Program on the Garigliano Nuclear Reactor, Quarterly Report No. 8, " July-September, 1964.
9. GEAP-4768/EURAEC-1316, "Development Program on the Garigliano Nuclear Reactor, Quarterly Report No. 9, " October-December, 1964.

Topical Reports

1. GEAP-4423/EURAEC-1055, "Reactor Pressure Vessel Material Surveillance Program at the Garigliano Nuclear Power Plant," F. A. Brandt and I. R. Kobsa, March 1964.
2. GEAP-4356/EURAEC-1212, "Physics Specifications for SENN On-Line Computer-Preliminary Report," N. H. Barth, July 1964.
3. GEAP-4702/EURAEC-1313, "Specification of SENN On-Line Computer System and Thermal Hydraulic Calculational Functions," by T. Sorlie, October 1964.

APPROVALS

C. L. Howard
C. L. Howard, Project Engineer

S. Levy
S. Levy, Manager
Heat Transfer and Reactor Projects

D. H. Imhoff
D. H. Imhoff, Manager
Engineering Development

REFERENCES

1. W. R. Morgan "Power Distribution Measurements at Dresden", Trans. ANS, to be published.
2. M. L. Batch, et al., "Mid-Life Power Distribution in the Indian Point Reactor", Trans. ANS, Vol. 7, No. 2 (494), November 1964.

DISTRIBUTION

No. of Copies

U.S. Atomic Energy Commission 2111 Bancroft Way, Berkeley 4, California	
Attn: W. H. Brummett, Jr., Director, Contracts Division	23
Attn: Director, Reactor Division	1
U. S. Atomic Energy Commission Division of Reactor Development Washington 25, D.C.	
Attn: Chief, Water Reactors Branch Civilian Reactors	1
Attn: Chief, Reactor Engineering Branch Naval Reactors	1
Attn: Chief, Research and Development Branch Nuclear Safety	1
Attn: Asst. Director, Office of Foreign Activities	1
U. S. Atomic Energy Commission Chicago Operations Office 9800 South Cass Avenue Argonne, Illinois Attn: Director, Reactor Program Division	1
Societa Elettronucleare Nazionale (SENN) Direzione Generale in Roma Via Torino 6 Roma, Italy	1
International General Electric Operations, S. A. Casella Postale No. 5 Scauri, Latina, Italy Attn: D. L. Layton	1
International General Electric Operations, S. A. 42-44 Niddastrasse Frankfurt/am Main Germany Attn: V. G. Grayhek	1
Westinghouse Electric Corporation Bettis Atomic Power Laboratory P. O. Box 1468 Pittsburgh 30, Pennsylvania Attn: Library	1
Knolls Atomic Power Laboratory P. O. Box 1072 Schenectady, New York Attn: W. Cashin	1
Argonne National Laboratory Argonne, Illinois Attn: Robert Macheray	1

DISTRIBUTION (Continued)

	<u>No. of Copies</u>
Northern States Power Company Minneapolis 2, Minnesota Attn: D. F. McElroy	1
General Nuclear Engineering Corporation P. O. Box 245 Dunedin, Florida Attn: J. M. West	1
Allis-Chalmers Manufacturing Company Atomic Energy Division Milwaukee 1, Wisconsin Attn: H. Etherington	1
Allis-Chalmers Manufacturing Company Nuclear Power Department P. O. Box 8697 Washington 11, D. C. Attn: H. C. Nickel	1
Consumers Power Company 212 W. Michigan Avenue Jackson, Michigan Attn: R. E. Kettner	1
Babcock and Wilcox Company Atomic Energy Division Lynchburg, Virginia Attn: J. F. Mumm	1
Division of Technical Information Extension P. O. Box 62 Oak Ridge, Tennessee Attn: R. L. Metter	3