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MINUTES OF MEETING
PWR
SUBCOMMITTEE OF THE ACRS WITH WESTINGHOUSE

Bettis Field, October 10, 1955

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Present for Westinghouse:

S. Krasik
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J. E. Nolen
N. J. Palladino
C. M. Shapiro

Subcommittee:

I. B. Johns, Chairman
D. A. Rogers
R. C. Stratton
Irving Kaplan
H. Wensel, Secretary

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The meeting was opened by Dr. Krasik. He stated that, although the design of the PWR is not complete enough for a safety report to be written, it is hoped that a reasonably complete report can be prepared by December 1. The purpose of the present meeting was to outline the proposed content of the report and to get suggestions from the subcommittee on additional topics that may be required. The preliminary report on the PWR, written by John Simpson for presentation at the Geneva Conference, will be distributed to the members of the ACRS.

The agenda of the meeting included (1) Over-all Plant Design, (2) Reactor Design, (3) Reactor Physics, Normal Protection and Anticipated Accident Conditions, (4) the Containment Problem, and (5) Outline of the Report.

1. Over-all Plant Design. Mr. Ellis

The site is protected from flood by improved flood control upstream and by its elevation, approximately 70 ft. above normal stream level. It was noted that all cooling water must be pumped up about 55 ft.

The meterology and hydrology of the site will be studied since it is known to be different from that at the Bettis site.

The reactor is designed to permit removal of the entire core, though Dr. Krasik felt that unloading will probably be done sectionwise. It was mentioned that the pool water, into which the core is moved could be poisoned, for

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safety, though no definite decision has been made regarding method of unloading. It was emphasized that the ACRS would be very critical of the proposed procedure.

The worst accident is considered to be rupture of the primary loop. This will lead to a pressure rise in the vapor container of about 52 psi. The vapor containers which enclose the entire reactor and steam system are designed for this working pressure. These containers are all interconnecting through very large (8 foot and 12 foot) ducts and are protected inside by a 6-inch layer of "Gunnite" protection against missiles and thermal shock.

It was pointed out by Mr. Rogers that pressure vessel codes do not necessarily apply to such large vessels carrying such large pipe penetrations. He suggested that an independent analysis of the vapor containers should be made. He also suggested that the design, in its broad features at least, be submitted to the ASME for their opinions regarding the applicability of their code. Dr. Krasik stated that the State of Pennsylvania has already approved the design of the PWR vapor container.

Penetrations into the vapor container for electrical connections are well designed. Ventilation is provided by two 44" ducts fitted with butterfly valves actuated on signals for pressure, temperature and radiation. These valves are designed for 60 psi. The containers will be tested to 70 psi. cold and are designed to lose not more than 1/10% in 24 hrs. at 52 psi. The design of the closing mechanism for the butterfly valves is not complete. The design will be discussed in the report, along with the consequences of escape of some radioactive steam into the building resulting from slow closure of the valves.

II. Reactor Design. Mr. Palladino

The reactor will contain 52 Kg U^{235} in flat zircaloy clad enriched "seed" plates and 11.6 tons natural uranium as UO_2 canned in tubes. The control mechanism is being designed, but details are not yet available. This topic will be treated in the report. Water flow through the reactor will be orificed to match power distribution. This distribution is expected to be very uneven.

The UO_2 pellets are pressed and sintered to 93% of theoretical density, ground to size and canned in zircaloy tubes. Center temperatures reach 2,500°F but irradiation tests run at fluxes considerably above the operating level expected in this power plant have resulted in no physical damage.

Loss of coolant flow will result in steam formation in 1 sec. and in 3 sec. the tubes will have 40% quality steam. The maximum power density in the blanket is expected to be about half that in the seed. The temperature coefficient of reactivity is determined principally by the seed region and is at least $2 \times 10^{-4}/^{\circ}F$ and probably 4×10^{-4}.

In spite of plutonium build-up, reactivity will decrease with time of operation. The reactor will probably not go critical with two adjacent control rods lifted. In the hot condition the rods will shut down to -10 to $-15\% \Delta k$.

III. Reactor Physics and Accident Appraisal. Dr. Krasik.

Scram will be initiated by neutron level, flux period, temperature, and coolant flow. The following types of accidents will be analyzed: Rod withdrawal, cold water, loss of coolant flow, and primary loop break.

1. Rod Withdrawal Accident

It is expected that no incident due to rod withdrawal can cause melting of the fuel plates. The temperature coefficient of the PWR is twice that of the STR and for periods down to 10 milliseconds in the STR the reactor is stopped by boiling in the channels before fuel plates can melt. The reactor is protected by having four 100 c sources and complete overlap of instrument response from zero to full power. A serious accident due to uncontrolled rod withdrawal seems remote.

2. Cold Water Accident

The water loops will probably be controlled by temperature sensing devices which control the valves primarily to avoid thermal shock. This will prevent a cold water surge from putting the reactor on a period shorter than 10 sec. Dr. Krasik considers that a cold water accident is not likely to be serious, but will give a thorough analysis of it in the report.

3. Loss of Coolant Flow

Scram is initiated by loss of power to the pump and also by loss of pressure drop in the system. It is estimated that if power to all pumps is interrupted and the controls start to move within 1 second, the coasting of the pumps will prevent boiling in the hottest channel. Some boiling can be tolerated. This situation will be treated in the final report.

4. Rupture of the Primary Loop

For a small break (1/2" diam.) the charging pump can maintain coolant pressure, and reactor can be scrammed, isolated by valves, and allowed to cool by natural convection. The temperature rise will remain within safe limits.

For a larger break (2-3" diam.) the pumps cannot make up the loss, but the core will probably remain under water and no plates will melt. At 1000-1200° C the reaction of zirconium with water is accelerated. If all the zirconium in the reactor (11 tons) reacts the energy release will be equivalent to the stored energy in the water, and pressure in the vapor containers will reach 70-75 psi. rather than 52 psi. due to flashing of water alone. However, the zirconium in the blanket is not expected to react.

An excellent study of the zirconium-water reaction was presented by Dr. Lustman showing that reaction is very incomplete and cannot be catastrophic. The subcommittee requested that this work be made available in detail. Dr. Krasik stated that it will be included in an appendix in the report.

A study is being made of ways for keeping the core covered with water in case of a major break. No satisfactory method can be recommended at present. Therefore the consequences of a major loss of water were reviewed. The zirconium reaction will give no explosive release of energy and reaction will be far from complete. The problem of handling the hydrogen liberated in the reaction is being studied but no solution is evident. This may be very serious. Cooling sprays are being considered, but an inert atmosphere (nitrogen) in the vapor containers is not possible because of the need for ventilation for cooling and maintenance operations.

The problems of how to handle the gaseous contents of the vapor containers after an incident is a serious one that requires much more study. WAPD is anxious to study any and all possible methods.

5. <u>Outline of the Report.</u>

Dr. Krasik listed the following topics for the summary report in December:

1) Purpose, scope and general philosophy of the project.

2) Description of the site and surroundings, including seismology; meteorology and hydrology cannot be completed in the time available.

3) Description of the plant - fairly complete but not detailed.

4) Primary coolant water chemistry - complete.

5) Primary loop materials of construction, inspection, testing - complete.

6) Reliability of the primary loop.

7) Design of vapor container - complete. Westinghouse has asked Stone and Webster to analyze the reliability of the vapor containers, but do not expect to have a report completed by December.

8) Waste disposal system - preliminary descriptions only.

9) Organizational responsibilities of the AEC, Westinghouse, Duquesne Light, and the contractors will be described and also the codes under which the plant will be operated.

10) Analysis of escape of radioactivity and atmospheric diffusion under normal operation and for various malfunctions short of rupture of the vapor container.

11) Analysis of accidents and the protective measures to be taken. This will include failure of equipment, break in primary loop, start-up accident, cold water accident, etc.

12) An appendix giving the results of studies on the metal-water reaction and the hydrogen problem.

Members of the subcommittee suggested the following additional studies be made and included in the report.

a) Detection of fuel element failure and methods for handling the problem.

b) Effects of radiation on the properties of the metals of the primary loop as well as the vapor container.

c) Analysis of the validity of applying present codes to such large vessels.

d) Analysis of the worst accident <u>with</u> rupture of the vapor container.

e) Description of loading and unloading procedures with associated hazards.

The subcommittee feels that Westinghouse is doing as thorough and fine a job as is possible under the se<u>ver</u>e time limitation. The status of the project, with its shortcomings, was presented in a completely frank and factual manner. It is anticipated that the major studies will be completed by December and that the remaining problems can be solved in time, as construction at the site proceeds.

This detailed report of the meeting is written to aid the Advisory Committee in appraising the problems when the summary report is received. Westinghouse wants a second meeting with a subcommittee about November 15 to review the material for the summary report. Members of the Advisory Committee are urged to send comments and questions to Dr. Wensel before that time so they can be mentioned to Westinghouse before the final draft of their report is written.