August 19, 1964

MEMORANDUM

TO : Dr. J. C. Ennis, Subcommittee Chairman

FROM : E. P. Fraley, Executive Secretary
ACRS

s/RHW for

SUBJECT: DRAFT MINUTES OF SAFETY RESEARCH (LOFT) SUBCOMMITTEE MEETING OF AUGUST 10, 1964

Attached for your review are the draft minutes of the subject meeting. Copies have been sent to remaining members of the Committee for information and/or comment.

cc: Chairman of ACRS
R. Duffey

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8/18/64

MINUTES OF
REACTOR SAFETY RESEARCH SUBCOMMITTEE MEETING
ON
LOFT
HELD IN WASHINGTON, D.C.
August 10, 1964

Purpose: The purpose of this meeting was to review the safety aspects of the proposed loss of coolant test at the LOFT facility.

Attendees:

ACRS

Dr. M. J. C. Knuth
Dr. D. Okrent
Col. R. C. Stratton
Dr. L. Silverman
Mr. M. J. Palladino
Mr. W. D. Manly
R. F. Fraley - Executive Secretary

Division of Reactor Licensing

R. G. Cano
F. B. Anderson
R. L. Tedesco
J. J. Zino
R. R. Meservey
R. L. Waterfield

AEC - Idaho Operations Office

W. [illegible] Connell
R. S. Wright
G. Watson

Division of Reactor Development

J. A. Lieberman
M. Booth
T. G. Schleiter
Edward Gilbert
R. H. Davidson
T. E. Alsaga
A. J. Prassesky
S. A. Szawlewicz
H. L. Bemester
H. Pembroke

Phillips Petroleum Co.

J. M. Waage
O. M. Hauge
T. R. Wilson

Kaiser Engineers

L. F. Carpenter

U. S. Weather Bureau (AEC/DRD)

I. Vander Hoven

Executive Session: Dr. Kouts noted that it appeared appropriate to comment on the programmatic as well as the safety aspects of the Loss of Coolant (LOC) test. He suggested that letters on both subjects might be appropriate following the ACRS meeting.

Dr. Silverman expressed concern over the lack of any engineered safeguards which are adequate to compensate for having a single containment barrier after a deliberate MCA. He noted that the building spray system requires about four hours to reduce the containment pressure to atmospheric. The air cleaning system cannot be started until the pressure is relieved since it is a low pressure system. He suggested redesign (e.g., increased capacity of the building spray system) or additions of other engineered safeguards (e.g., an internal foam system) which could be used to rapidly terminate fission product releases to the environment if the containment should fail.

Dr. Kouts noted that it is very unlikely that meteorological conditions of the type required will last for a 4-day test period as planned. A change in atmospheric conditions could require termination of the test with a resultant loss of a year's time and effort.

Meeting with Regulatory Staff: E. Case reported that the Staff is still evaluating several unresolved areas of concern. These involve items listed below. In addition, he noted that DRL has not yet concluded whether Part 20 or Part 100 dose limits should be used in judging this test.

2. Adequacy of the planned MCA evaluation.

(a) Fission-product release model - A time dependent fission-product release model is used by Phillips Petroleum Co. in their hazards evaluation. An instantaneous release of fission products could give a greater source term by a factor of 30.

(b) Meteorological diffusion parameters - A more pessimistic set of parameters, which could well develop over a 4-day period, could give increased doses by a factor of 10^2 - 10^3.

(c) Leak Rate Assessed - The measured containment leak rate could be a factor of 1.5 too low when one considers the uncertainties involved in thermal effects, steam-air mixtures, etc.

2. Consequences of an unplanned MCA.

(a) The transportable nature of the experiment reduces the missile shielding provided. (The doors provided for experiment entry are not provided with missile protection).

(b) A start-up accident has not been evaluated in detail.

Dr. Ehrenk asked if the dolly which holds the reactor could be overturned by reactive forces during coolant blowdown. This could change the planned MCA to an unplanned accident. The forces involved are apparently large enough to overturn the dolly. A balanced rupture is planned, however, and will be checked during the non-radioactive blowdown tests.

Dr. Knuth mentioned that the scope of tests in the Containment System Experiment facility has been scaled down to the point where consideration should be given to development of information concerning performance of engineered safeguards in the LOFT program.

Meeting with IRD and Contractors: Dr. Lieberman reported on the status of the SNEP program as follows:

NRFP: Shakedown runs in progress. Irradiated samples to be run in September.

SPERT-III: FL-2 fuel (UO_2) kinetics tests in approximately one month.

SPERT-I: Starting to operate as a transient capsule driver. UO_2 core installed.

SPERT-IV: PWR critical tests in progress. SPERT-I core will be transferred to SPERT-IV in the near future. SPERT-I will then be shut down.

SPERT-II: Shutdown, gas cooled transient test core being evaluated.

CSE: Containment vessel contractor selected (Chicago, Bridge & Iron). Invitation to bid on the pressure vessel has been distributed.

Dr. Knuth inquired if radioactive materials would be used in CSE to test the

value of various engineered safeguards.

Dr. Lieberman noted that only small quantities of active material are to be used because of the difficulty in cleaning up the facility. One must choose between many tests with low activity or a few tests with high activity. This subject may be discussed at a future date with the Safety Research Subcommittee.

Mr. R. Wilson summarized the objectives of the loss of coolant (LOC) test in [illegible]. This test will:

1. Demonstrate the consequences of a LOC accident in a PWR and the ability of the containment to retain fission products.

2. Develop quantitative information on the reactor and components and on fission product behavior.

3. Identify the phenomena of primary importance in accident consequences. Analytical studies have been in progress for more than a year to provide a framework for design and identify key factors.

Five phases will be involved, namely:

I. Containment leak tests.
   (a) Air leak rates.
   (b) Stresses.

II. Non-nuclear blowdown.
   (a) Variables: leak size, location, system pressure, temperature.
   (b) Mechanical behavior.
   (c) Thermal behavior.
   (d) Transport phenomena (out of the reactor pressure vessel).
   (e) Coolant state and location.
   (f) Leakage.

III. Operation to build in fission products (400-1800 hours)

IV. Loss of Coolant.
   (a) Nuclear power behavior
   (b) Core thermal behavior
   (c) Fission product releases
   (d) Fission product transport and plateout.

(e) Temperature-time behavior.
(f) Fission product behavior.
(g) Retention in steam/water.
(h) Divert radiation.

V. Post-test evaluation.
(a) Fraction of clad failure.
(b) Final fuel configuration.
(c) Fission product retention in fuel.
(d) Plant component damage.
(e) Fission product plateout and physical state.

The test will not be an upper level determination but will be in the range of the upper limit and will be used to tie significant points to other areas of safety R&D research. The Hazards Analysis, however, is based on what Phillips considers conservative estimates. Key questions to be answered are the core thermal behavior and fission product transport.

Mr. Kouts questioned the accuracy with which leakage rates can be determined when a condensing, steam-air mixture is present in the containment. Mr. Hauge agreed that development of an accurate method will require additional R&D.

The fission product release model and the accuracy of blowdown measurements were discussed briefly. It was noted that the core melt down model being used is the same BAW model proposed for the N. S. SAVANNAH time-to-melt calculations. Development of a two-phase flow measuring device is being explored.

Mr. Silverman commented on the limited engineered safeguards being provided and inquired why a core spray system has not been included. IREP representatives replied that the effects of a core spray would alter the natural sequence of events from a pipe break which they hope to investigate. Mr. Silverman noted, however, that spray systems are provided on several large reactors and that LOFT would provide an opportunity to obtain information on the performance of this engineered safeguard. He suggested that flow tests during Phase II would at least provide information on flow rates and distribution. T. R. Wilson agreed that this suggestion would be considered. Sizing and operation of the containment spray system in a manner more representative of an engineered safeguard was also discussed. M. Hembree again stated that operation was planned only to terminate the test since it would upset the natural balance of fission product distribution. A small recirculating air-cleaning system was suggested as a test of another engineered safeguard. The IREP representatives again commented that this could alter the desired test results.

Phase V will consist of hot cell examinations in the TAN area. Detailed analysis of data will be used to correlate LOC test results with associated safety R&D programs.

Mr. Palladino asked about the NDT of the containment vessel. The NDT of this vessel is about 10° F. The temperature in the winter, when the test will be run, is frequently down to minus 40° F.

Pipe penetrations, electrical penetrations and the sampling scheme were briefly described.

Following lunch several specific questions were discussed. These are summarized briefly below:

Q. Why is scram delayed until blowdown is complete?

A. To observe the effect of void formation on turning the transient.

Q. With what degree of dependability can meteorological conditions be predicted over a 4-day period?

A. Prediction for the 1st day is good (70 - 80% accuracy).
Prediction for the 2nd day is fair (50% accuracy).
Prediction for the following days is poor.

It was noted, however, that weather records indicate 4 - 5 day periods over which the weather remains reasonably constant. If unfavorable conditions were to develop the test would have to be terminated.

Q. Is relief valve capacity limited to 5% of reactor power as implied in the HSST?

A. Relief valve capacity has not been firmed up. It will be in accordance with Section III of the ASME Code.

Q. Will use of a carbon steel reactor vessel produce problems with Crud?

A. Based on experience at ANL and several carbon steel loops, excessive crud buildup is not anticipated. A factor of 5 on crud level is expected, however.

Q. Why is S.S. clad to be used? Most reactors are planning a return to Zr clad.

A. S.S. clad was based on a survey of power reactors some years ago. The containment is designed to withstand a metal-water reaction in the event Zr clad is used.

Q. Why is a ventilated containment used rather than a scheme with a closed recirculating-cooling system?

A. (The designers had apparently not considered this type of system).

Q. Are vital items powered routinely from the emergency bus only?

A. These items will automatically transfer to commercial power supply if the emergency bus fails.

Q. Reactor control from secondary pressure is an unusual arrangement. Why was it selected?

A. The pressure signal controls louvers on the air condensers. This results in reactor control via the temperature coefficient.

Q. There appear to be few process instruments for reactor safety. It was suggested that the plant design should conform to conventional practice.

A. Several controls such as core ΔP, ΔT and core outlet temperature actuate alarms. Either high or low reactor outlet pressure causes a scram.

Q. Has insulation been considered on the containment vessel to maintain a vessel temperature above the NDT?

A. This is not planned but it could be done.

Q. No safety injection appears to be installed.

A. Only the normal 5 gpm charging system is provided for addition of water.

Q. Would a LOC test with adequate fission product inventory to heat but not melt the fuel be helpful in planning for the melt down test?

A. It would be helpful but cannot be justified because it would require replacement of the core and instrumentation to eliminate the effects of possible thermal/mechanical damage.

Q. Would a melt down experiment smaller than the LOC test but representative of LOC test conditions be helpful?

A. Yes, but it is not considered necessary since this experiment is planned to establish the important parameters. Additional tests are planned in the LOFT facility with additional water cores.

The safety evaluation was then discussed. Special provisions in LOFT to control the tests and fission products consist of:

1. An installed decontamination system.
2. Connections to collect and monitor penetration leakage.
3. A press-reduction system.
4. An extensive sampling system.
5. A melt down pan below the reactor.
6. Balanced blow down nozzles.
7. Short reactor operating time.
8. A 6-mile exclusion area.
9. A downwind monitoring system.
10. An underground operating area with a self-contained breathing air supply.

Radiation doses calculated by Phillips give values of:

	Whole Body	Thyroid
Av.	0.17 rem	0.5 rem
Max.	0.5 rem	1.5 rem

These doses are based on a time-melt model which releases iodine fission products in the following manner:

Time	Transport Mechanism	% Released
0 - 1.5 hours	Vaporization of water	1.6%
1.5 - 4.0 hours	Bottom head heats	10.4%
> 4.0 hours	Bottom head melts	28 %
		40 % Total

It was noted that several things, such as an oxidizing atmosphere, could increase the rate of fission product release.

Phillips has not calculated the results of a TID-14844 release. A release consisting of 100% of the iodine would give 0.7 rem at the Naval Reactors facility (20 mi) and 1.0 rem at the site boundary (6 mi) if one assumes poor meteorology. If an inversion condition is used a dose of 10 rem (thyroid) occurs at the site boundary from a 100% iodine release.

Dr. Koutz noted that several factors used in the dose calculation required resolution with DRL before the ACRS meeting. He listed the following are for discussion at the August ACRS meeting:

Accident assumptions - melt down model.
Reactivity accidents - assumptions and consequences.
Containment ECCT problem.
Larger containment spray system.
Reactive forces acting to overturn the dolly.
Missile shielding and damage.

The meeting adjourned at approximately 4:45 P. M.

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