ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

October 28, 1963

MEMORANDUM

To : All ACRS Members

From : R. H. Wilcox, Assistant to Exec. Secretary
ACRS
RHW

Subject: REPORT OF CATEGORY B, C & D DOCUMENTS - SUPPLEMENT NO. 45

The attached list, Supplement No. 45, includes all Category B, C and D documents received in the ACRS office from September 21, 1963 through October 18, 1963.

Time will be provided at the 51st meeting to discuss the items indicated.

Please let us know if you desire to see copies of any of the reports listed.

For the benefit of the new ACRS members, the definition of documents as classified by this office is summarized below:

Category A - Forwarded for ACRS action - 18 copies received - distributed to full ACRS. In general, these are the principal hazards reports and application amendments to be reviewed and reported on by the Committee. These are sent to members upon receipt and without any special notation.

Category B - Forwarded for ACRS information but pertaining to projects on which the ACRS has previously advised - 3 copies received. These are evaluated by the ACRS office and sent to the Subcommittee Chairman and ACRS Chairman with a covering "Category B memorandum". These are also summarized periodically on the "Category B, C & D List" sent to all members. Subcommittee Chairmen review these documents and are responsible for recommending a course of action to the Chairman. Appropriate Category B items are discussed at ACRS meetings.

Category C - Forwarded for ACRS information - involves projects on which no ACRS advice will normally be requested (i.e., low-power University reactors) - 3 copies received. These are summarized on the "Category B, C & D List" and distributed to members on request.

Category D - Forwarded for ACRS information - these are reports on R&D work, which is applicable to projects in general. These reports are summarized on the "Category B, C & D List"; 3 copies are received and are available on request.

Reports on Safety Research received from the Division of Reactor Development Nuclear Safety group (J. Lieberman) are provided in 18 copies and distributed to the full Committee. ~~These reports are also summarized on the Category D list issued by this office~~. Reports in this Category ~~will~~ be identified in the future by ~~an appropriate stamp to distinguish them~~ from Category A items. or stamped "Reactor Safety Research Document"

Attachment:
Supplement No. 45

CC: D. Duffey

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SUPPLEMENT NO. 45 CATEGORY B October 28, 1963

CVTR

DL&R has issued Change No. 3, dated 9/28/63 to the CVTR Technical Specifications. This change approves alterations in the administrative organization and in the safety committee as requested by the licensee.

This proposed change was discussed in Category B reports dtd 7/22/63 and 9/19/63 and with DL&R at the 49th meeting. No ACRS action appears warranted. (To JCG & DBH - 10/9/63).

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CON ED
(INDIAN POINT)

A Con Edison letter dtd 10/7/63 requests a modification in the cold water accident prevention interlocks at Indian Point.

Presently, a main valve cannot be opened unless the loop temperature is within 20°F of reactor temperature, and pumps cannot start unless the $\Delta T$ is less than 35° F. The licensee would add an interlock which insures that the reactor is shutdown before an idle loop is started, and would increase the $\Delta T$ setting to 100° F. The proposal would permit a pump to be started with valves open and the reactor running, but only if the $\Delta T$ is less than 35° F.

The proposal will actually speed up return of a loop to service. Con Ed has looked into the physics considerations and has concluded that an adequate shutdown margin will be maintained when a loop is cut in.

No significant safety problem appears involved, hence no ACRS action seems warranted.

Con Edison ltr dtd 10/4/63 transmits "Survey of Environmental Radioactivity in the Vicinity of Indian Point Station, February 1 through July 31, 1963" dtd 8/30/63.

Atmospheric fallout is reported to be still the dominating influence in the environmental radiation background.

No ACRS action appears warranted. (To KRO & DBH-10/22)

SUPPLEMENT #45 CATEGORY B

CON ED
(INDIAN POINT)

Con Edison Company Semi-Annual Operations Report No. 2, dtd 8/30/63, describes Indian Point operations for the period February 1 to July 31, 1963. A total of 543.4 x $10^6$ gross Kw-hrs was generated during this period.

Fifty-five plant shutdowns were experienced, of which 24 were scrams and the remainder fast insertions. All such shutdowns are described in the report, but there appears to be no significant safety problem indicated here, and in fact, this represents a tremendous reduction from the 187 shutdowns in the previous reporting period.

No clad failures have been detected. There is a primary to secondary leak in one steam generator which has resulted in radioactivity release through the blow-down from this unit. When diluted with circulating water, this is very small, however, in fact, averaged circulating water outlet activity appears to be running less than inlet. Two accidental incidents resulted in the release of 34/!C to the sewage treatment plant, but procedures have been revised to prevent recurrence.

A number of equipment difficulties were experienced, some of which forced a reduction in station output, but were only peripherally connected with safety. These were:

1. A circulating pump tripped out due to a failure in the motor stator.

2. Two primary coolant pumps experienced gradually increasing bearing temperatures and were removed from service.

3. Superheater tubes around the burner throats have ruptured.

4. A heavy run of striped bass has caused some blockage of the circulating water screens. A large net is being tried to solve this problem after two other proposals failed to help.

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SUPPLEMENT #45 CATEGORY B

CON ED (INDIAN POINT) - Cont'd.

5. One generator gland seal has leaked hydrogen. (Other monitoring for hydrogen has not found any leaks.)

6. Copper-bearing pipes in the vicinity of the air ejector monitor have corroded badly.

7. A superheater inlet valve bonnet developed a severe steam leak which resulted in some cutting of body and joint.

8. Many rupture discs have failed and have been replaced with discs of modified design.

9. The cement lining of the circulating water lines has broken off and pieces have obstructed some of the condenser tubes.

All of the above have been repaired or are undergoing corrective action.

Since March there has been a constant disparity among the power level ion chamber readings, which is thought to be due to power distribution assymetry. This is still being investigated, however.

Containment leak testing shows an apparent negative leak rate, which Con Ed believes is due to in-leakage from a closed system (such as compressed air) into the sphere.

No significant safety problems are evident in this report, and no ACRS action therefore appears warranted. (To KRO & DBH - 10/2/63).

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DRESDEN

By ltr dtd 9/17/63, Commonwealth Edison Company repeats an earlier request for a change in the requirements for control drive and blade testing.

This proposed change was discussed in a Category B report dtd 9/23/63, and no ACRS action appears warranted.

By another ltr dtd 9/17/63, Commonwealth proposes to increase the heat flux limits and decrease the minimum burnout ratio to be used at Dresden.

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SUPPLEMENT #45 CATEGORY B

DRESDEN (Cont'd)

The licensee would like to operate at a minimum burnout ratio of 1.5, based on the Levy correlation (APED-3892). This would be consistent with burnout ratio limits previously approved for Big Rock Point and Humboldt Bay.

Commonwealth would set the heat flux limits for Type I, II, and III fuel at values equivalent to about 15.2 Kw/ft. at rated power, as compared to present limits of 14 Kw/ft. Toward the end of core life, present limits would be exceeded, and the new limits approached. The licensee claims that recent $UO_2$ conductivity information indicates that these limits would not result in center melting (but just barely) at 1.25 times full power. Tables of irradiation data are included to support the proposed increases.

The proposed limits do appear to represent decreased safety, and, in particular, the heat flux limits appear to be beyond those previously accepted by the ACRS for any project. The irradiation data include some failures. Discussion of these limits with DL&R at the 51st meeting therefore appears warranted. (To HWN & DBH - 10/15/63).

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GETR

General Electric Company ltr dtd 8/28/63 informs the AEC of plans to irradiate "pseudo-stable materials" (explosives) in the GETR trail cable facility.

The Aerojet General Corporation wishes to irradiate capsules containing less than 2 grams of PETN, TNT, HMX/EXON, DATB or ANATOL. The capsule is designed to contain an accidental detonation with a double load of explosive. Capsule tests are being conducted out-of-pile, and to date these have been quite successful. Detonation in-pile is not planned and not expected.

General Electric apparently believes that the GETR license will permit these irradiations, and plans to proceed if out-of-pile testing discloses no problems. The proposal does not appear to involve significant hazard, hence no ACRS action appears warranted. (To LS & DBH - 10/21/63).

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SUPPLEMENT #45 CATEGORY B

GETR

General Electric Company ltr dtd 10/10/63 transmits Proposed Change No. 2 for the GETR.

The licensee proposes to conduct a "critical heat flux" test in the GETR boiling water loop. A four rod fuel element will be installed in the loop, and coolant flow will be gradually reduced until all the rods are taken through the critical heat flux point (DNBR=1) and into film boiling. Unless a failure occurs, operation would continue for about ten minutes under these conditions.

The fuel will be $UO_2$, and the clad on two of the rods will be Zircalloy, the other two stainless steel. The $UO_2$ pellets will be cored with a center hole to reduce the likelihood of $UO_2$ central melting.

GE has done a rather thorough analysis in preparation for this test. Cladding temperatures are expected to go up about 1000° F (to 1600° F) upon initiation of film boiling. Bending stresses in the clad will exceed yield; blisters may form and become unstable; fracture of the cladding may follow, however GE does not think so.

It is obvious that this test involves some hazards. The emergency cooling system will be disconnected. Fuel would most certainly fail if there were a loss of flow or pressure in the loop during the test.

On the other hand, GE will take a number of precautionary steps, including the reduction of loop leak rate, minimizing fission product inventory, providing storage tankage for loop coolant, providing special handling equipment, and conducting a shakedown run with an unfueled, mock-up test section. The results of the test should be very valuable in the understanding of burnout hazards in water cooled reactors.

GE states that the consequences of a fuel element failure would be less severe than some which have occurred at GETR in the past. The consequences of a loop maximum credible accident would be less than those previously analyzed. With everything considered, it would appear that no ACRS action is warranted. (To LS & DBH - 10/22/63).

SUPPLEMENT #45 CATEGORY B

HALLAM

A TWX from Atomics International dtd 9/24/63 reports that control rod thimbles of 304 stainless steel are being substituted for all zircaloy thimbles in the HNPF.

Data on HNPF control rod performance in stainless steel thimbles has consisted of a minimal number of tests at room temperature at the HNPF and additional tests, including some at temperature, with a Hastelloy-X poison column which are underway at AI's Santa Susana test tower. The latter have indicated no binding or sticking, but some scratching of the thimble by the rod. An AI senior metallurgist believes these are surface defects and not galling, however.

AI calculates that the decrease in available core reactivity due to the change in thimbles will be 200 cents. While the statements are not very definitive, it appears that the decrease in total rod worth will be about the same.

Thimble atmosphere will be maintained at 15 psig, and a gas chromatograph will be used to monitor for such gases as hydrogen. A hydrogen removal device can be started if hydrogen is found in the thimbles.

Trouble has been experienced with leakage of hydraulic fluid from the control rod snubbers. Some O-ring seals are therefore being removed and replaced by seal welds.

The installation of stainless steel thimbles was discussed at the 49th and 50th meetings. (To RCS & DBH 10/2/63).

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HALLAM

A ltr from Atomics International, No. 63AT6472 dtd 9/26/63, confirms the information previously provided by TWX (See Category B report dtd 10/2/63) and provides additional data concerning the replacement of zircaloy thimbles with stainless steel at Hallam.

The control rod drop tests at AI's Santa Susana tower are now complete, and there has been no evidence of galling, sticking, or binding after 1800 four foot shim cycles and 180 full scrams. The rod and thimble from this test are still to be finally examined, however.

Drop time data for two rods in stainless steel thimbles at the HNPF is also reported, and no difficulties are apparent. Prior to reactor start-up, each rod was to be drop tested.

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HALLAM (Cont'd)

AI also proposes to eliminate the piston ring seals which were located at the lower end of the thimbles. This will leave a 10 mil clearance between each thimble piston and the lower grid plate. AI has submitted a calculation of the sodium leakage through these clearances, however, and this comes to only 0.7% of total reactor flow.

AI has also submitted a drawing which shows the location of the seal weld on the shock absorber assembly. This weld is intended to stop leakage of hydraulic fluid through this seal.

The use of stainless steel thimbles was discussed at the 49th and 50th meetings, and no further ACRS action appears warranted. (To RCS & DBH - 10/3/63).

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HALLAM

An Atomics International ltr dtd 10/1/63, 63AT6287, proposes numerous changes in the Technical Specifications for the HNPF. The following appear to be the most significant proposals, from a safety standpoint:

1. A relaxation of the requirement that negative pressure be maintained in the reactor building which would permit primary cells to be open along with equipment access doors, so long as the reactor has been down more than two weeks and no irradiated fuel is being handled.

2. A relaxation of the 1% limit on oxygen in the nitrogen atmosphere which would permit reactor operation up to 100 Kwt with essentially no oxygen limit.

3. An increase from 6 to 18 in. of water for the allowable primary cell pressure.

4. Modification of sodium valve interlocks to permit control rod withdrawal (except initially) with certain valves not fully open.

5. Reduction of minimum nitrogen inventory from 100,000 to 60,000 scf.

An Atomics International ltr dtd 9/27/63, 63AT6527, reports on modifications to the loading face shield cooling system in order to alleviate compressor overloading, pipe hammer, noisy check valves and excessive nitrogen leakage.

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HALLAM (Cont'd)

This system has been extensively redesigned, and all of the above problems appear to be solved.

An Atomics International ltr dtd 10/3/63, 63AT6624, reports the investigation results and ensuing modifications to the secondary sodium pumps as a result of the binding of all three pumps which occurred on 6/9/63.

The binding of all three pumps was due, at least in part, to the introduction of foreign matter into the pump bearings or other close running fits. An uneven circumferential thermal gradient contributed to the seizure of one of the pumps.

Three steps were taken to avoid similar troubles in the future:

1. The running clearances of the impeller upper and lower wear rings were increased.

2. A forced convection cooling system was added to each pump to minimize the temperature gradient.

3. Sodium filters were moved to a location which permits the sodium to be filtered as a loop is refilled. Additional by-pass filtering was also used to remove remaining metal particles.

Subsequent operation has apparently proved out these modifications. DL&R (Ireland) thinks that the flakes come from the spacer bars which were installed in the IHXs. He anticipates further seizures when the pumps are stopped, but this is not a safety problem. An analysis of the primary pumps has convinced AI that similar problems will not be encountered there.

Another AI ltr dtd 10/3/63, 63AT6625, replies to a DL&R request for more information on the HNPF heating and ventilating system deficiencies. The system has proven inadequate to cool the sodium pump drives, and the main building exhaust fan has not met specifications.

Plans call for the installation of "collar type" hoods to suck air from directly around each pump drive into the exhaust system. Additional fans and dampers will also be installed to permit air recirculation in the event of loss of high bay supply air. Exhaust duct openings in the high bay are also to be relocated to prevent hot air stratification. Exhaust fan drives will be altered to increase fan speed.

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SUPPLEMENT #45 CATEGORY B

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| HALLAM (Cont'd) | This submittal also includes analyses of reactor vessel temperature transients. Three regions of the vessel were analyzed, and thermal stresses were not found to be excessive.<br><br>The proposals to reduce the confinement requirements and oxygen limits appear to be potentially significant decreases in safety without much operational justification (AI October 1, 1963 letter). DL&R (Ireland) has indicated that its review of these items is still underway. Some discussion of these items with DL&R at the 51st meeting, therefore, appears warranted. (To RCS & DBH - 10/18/63). |
| HUMBOLDT BAY | By application dtd 9/11/63, the Pacific Gas and Electric Company has submitted Proposed Change No. 13 to the Humboldt Bay Technical Specifications.<br><br>The licensee proposes to complete the installation of and to operate a continuous containment leak rate detection system at Humboldt Bay. Provisions have been made to supply air to both dry well and suppression chamber. Graphite rupture discs will be installed in the vacuum breaker lines between the dry well and suppression chamber.<br><br>The dry well will be pressurized to 20 in. of water and the suppression chamber to 10 in. When required, air will be added to either chamber to restore the pressure, and this makeup will be metered and used to calculate leak rate.<br><br>The method of extrapolating to design pressure is open to question both on this and other projects. Also, there will probably be some errors which are difficult to account for and which may mask the desired leak rate measurement. Nevertheless, the proposed monitoring should certainly detect any large leaks which develop, and hence should contribute to safety. No ACRS action therefore appears warranted. (To LS & DBH - 10/15/63). |
| RADIATION EFFECTS REACTOR (RER) - LOCKHEED | By ltr dtd 9/26/63, LGD/168786, the Lockheed-Georgia Company has submitted proposed Technical Specifications for the RER.<br><br>No ACRS action appears warranted. (To DBH - 10/9/63). |

SUPPLEMENT #45 CATEGORY B

PIQUA

A ltr from Atomics International, #63AT6094, dtd 9/9/63, reports on a planned modification to the electrical system at the Piqua facility.

A transfer switch and associated circuit breakers, etc., will be added to enable the "diesel driven alternator" to supply the plant lighting load. This will meet recommendations of the diesel manufacturer that this equipment be periodically operated under load conditions.

In the event of a 480 volt power failure while the diesel is carrying the lighting load, the diesel will drop the lighting circuits and be available to pick up the emergency load. It is not clear, however, whether it will do this automatically as it would if it were not carrying the lighting load.

DL&R (Gregory) has now looked into this, and has advised that it will pick up the emergency load automatically. No ACRS action therefore appears warranted. (To RCS & DBH - 10/9/63).

PRDC
(ENRICO FERMI)

A TWX frm PRDC dtd 9/26/63 contains the results of the investigation into the failure of safety rod No. 4 to scram. (See Category B report dtd 9/24/63).

A leak was discovered in a small bellows which normally excludes sodium from the interior of the safety rod extension. Thus, while the licensee was conducting a test at 10 psig cover gas pressure, sodium leaked through the bellows and into a region around the latch rod where the temperature was low enough for the sodium to freeze. This prevented the latch rod motion which is needed to delatch the safety rod.

PRDC does not believe that this is indicative of any "generic fault" in safety rod design. Another extension was to be installed on No. 4 rod. Before returning the reactor to critical, the licensee proposed to withdraw all safety rods and hold them out for 24 hours, then scram them. Any further difficulty will be investigated and reported.

This proposed test would apparently not be done at sufficient cover gas pressure to cause trouble if additional leaks exist. Although normal cover gas

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SUPPLEMENT #45 CATEGORY B

PRDC (Cont'd)

pressure is low, it would appear that the bellows is a sort of "Achilles heel" whose failure on a number of rods could conceivably negate the safety system if cover gas pressure ever gets out of hand. DL&R (Ireland) informs me that the test described above was run and that the delatch times had increased slightly. This problem was discussed at the 50th meeting. (To DBH - 10/2/63).

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PRDC

A TWX dtd 9/20/63 from PRDC reports the failure of one control rod to scram when called upon to do so as part of the experimental program. The rod was successfully driven in automatically as part of the scram back up.

Investigation has showed that magnet current was not interrupted. The reactor is shut down and will not be started until the failure is understood and corrected

No ACRS action appears warranted. (To DBH - 9/24/63)

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PWR

Duquesne Light Company report DLCS 5000863 describes Shippingport operations during August 1963. The station accumulated 373 "EFPH" during the month.

There were no major problems which arose during August. There was one safety shutdown caused by trainee error. There was one inadvertent decrease in primary pressure to 1700 psi.

Trouble has been experienced with the 18 in. primary loop stop valves. Under certain pressure conditions, these valves will drift from the fully closed position. These valves have also failed to open when called upon to do so, and pressures had to be adjusted in order to open them. Evaluation is in progress, but it is thought that there is a malfunction of the flow limiting check valve which relieves the hydraulic valve bonnet pressure during valve opening.

One pressurizer relief valve leaked excessively and was valved out of service.

As a special test, morpholine is being added to the condensate and feedwater systems. An indirect effect of this is that non-deaerated water is being added to the primary system.

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SUPPLEMENT #45 CATEGORY B

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| PWR (Cont'd) | Radioactive iodine has increased in the primary system due to peaking-type plant operations, and has correspondingly increased in 1A, 1B, and 1D boilers.<br><br>No safety problems are apparent which are not being pursued by plant personnel. No ACRS action appears warranted. (To RCS - 10/8/63 w/cpy to DBH). |
| NS SAVANNAH | By ltr dtd 9/30/63, the Director of Regulation has approved the Joint Group's request to non-nuclearly heat up and to operate the NS SAVANNAH's primary system subject to two conditions:<br><br>1. At least one containment airlock door is to be kept closed when the primary system temperature exceeds 300° F.<br><br>2. A reactor operator previously certified by DL&R is to be present during control rod manipulations.<br><br>No significant safety problem appears to be involved. No ACRS action appears warranted. (To FAG & DBH - 10/3/63). |
| SAXTON | By application dtd 8/26/63, Saxton requested an extension of the expiration date for its Provisional Operating License until 12/31/63 to allow the AEC time to issue a Final Operating License.<br><br>By order dtd 9/25/63, DL&R extended the date as requested.<br><br>No ACRS action is warranted. (To CRW & DBH - 10/7/63). |
| VBWR | The General Electric Company has filed an application, dtd 10/7/63, for Proposed Technical Specification Change No. 2. The licensee requests an alteration in control rod inspection requirements.<br><br>Since the Mark II control rods are not dependent on the poison material for support, GE believes the inspection requirements can be reduced. GE proposes to examine at least half the rods every six months and all rods at least yearly.<br><br>This change appears not to involve significant safety considerations, hence no ACRS action appears warranted.<br><br>(Cont'd next page). |

VBWR (Cont'd)

A General Electric ltr dtd 9/30/63 contains the Report of Change, Test, or Experiment No. 54. This involves two new superheat fuel assemblies to be irradiated in the ESADE loop of VBWR.

Chief differences in these elements from those previously irradiated are in the geometry, method of spacing, and in the use of Hasteloy X cladding.

Operating conditions, according to GE, will be less severe than for previous tests. No ACRS action, therefore, appears warranted. (To DBH - 10/18/63).

VBWR

A ltr from General Electric Company, dtd 9/19/63, reports that there was a reactor scram and building isolation at VBWR on August 30, 1963.

The scram and isolation resulted from the introduction of steam into a drain tank which in turn caused a high stack activity trip. Four men in the building at the time left without receiving any exposure or contamination.

This appears to be a minor incident from the standpoint of safety, and no ACRS action therefore appears warranted. (To WKE & DBH - 10/1/63).

VBWR

DL&R has issued Change No. 6, dtd 10/8/63, to the VBWR Technical Specifications. As requested by General Electric, this Change authorizes replacement of the steam turbine driver for the No. 2 circulating pump with an electric motor.

DL&R has approved all associated Technical Specification revisions proposed by GE except for an increase in permissible reactor power level with only one pump operating. This one has been deferred.

These changes were discussed in Category B reports dtd 5/16/63 and 9/17/63. No ACRS action appears warranted. (To WKE & DBH - 10/10/63).

YANKEE

A ltr from Yankee Atomic Electric Company dtd 9/26/63 supplies Proposed Change No. 46 to the Yankee Technical Specifications.

The licensee proposes to make piping alterations which will permit the boric acid transfer pump to take suction from the safety injection tank and discharge to the low pressure surge tank (LPST) makeup line.

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SUPPLEMENT #45 CATEGORY B

YANKEE (Cont'd)

This change will permit the operators to blend borated water with demineralized water which is fed to the LPST. Thus makeup to the primary system from the LPST can be made to have about the same boron concentration as the primary system itself.

No significant safety problems are apparent, hence no ACRS action appears warranted. (To CRW & DBH - 10/21)

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YANKEE

By ltr dtd 9/4/63, Yankee submitted Proposed Change No. 45 to License DPR-3.

Yankee proposes insertion of stainless steel spacer tubes to act as guides for each in-core flux wire thimble. The tubes would be installed in each fuel element which will contain a flux wire in either Core III or Core IV. Yankee believes that it is "prudent" to install these to reduce wear of the thimbles from flow-induced vibration.

The Change appears to be a minor one, and the means which are being provided to hold these new tubes in place appear adequate. No ACRS action therefore seems warranted.

DL&R has transmitted approvals of eleven different Yankee Changes. These are tabulated below:

| Yankee Proposed Change No. | Description | Discussed in Cat. B Rpt. dated | Addl. Restrictions by DLR | ACRS Action Appears warranted |
|---|---|---|---|---|
| 32 | pressurizer vent line | 3/28/63 & 9/23/63 | no | yes* |
| 34 | safety injection valves | 6/20/63 | no | no |
| 35 | pressurizer spray valve | 6/20/63 | no | no |
| 36 | Core III and boric acid at power | (ACRS review & ltr. at 47th mtng.) | yes | no |
| 39 | maintenance instructions | 7/22/63 | yes | no |
| 40 | Hf & new Ag-In-Cd control rods | 8/9/63 | yes | no |
| 41 | low pressure surge tank vent valve | 8/9/63 | no | no |

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SUPPLEMENT #45 CATEGORY B

YANKEE

| Yankee Proposed Change No. | Description | Discussed in Cat. B Rpt. dated | Addl. Re-strictions by DLR | ACRS action appears warranted |
|---|---|---|---|---|
| 42 | increased number of flux wire positions | 8/9/63 | no | no |
| 43 | shim rod material change | 8/16/63 | no | no |
| 44 | spent fuel storage rack | 9/23/63 | no | no |
| 45 | spacer tubes | this one | no | no |

*Change No. 32 has been approved by DL&R and was discussed at the 50th meeting.

No further ACRS action appears warranted. (To CRW & DBH 10/4/63).

University of Nevada - Issuance of Facility License and Notice to the Office of the Federal Register with Hazards Analysis.

DL&R has issued Facility License No. R-91 to the University of Nevada, authorizing operation of the Atomics International Model L-77 solution-type nuclear reactor located on the campus of the University at Reno, Nevada.

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Aerojet-General Nucleonics ltr dtd 9/10/63 and Report AN-552 "Radiological Safety Procedures" dtd March 1962.

This letter supersedes earlier information submitted by the licensee in support of a request to irradiate certain chemical compounds in the presence of fissionable materials to determine reactor kinetics and g-values of various fissiochemical reactions.

AN-552 This handbook was written to guide AGN employees in their handling of radioactive and toxic materials and applies specifically to occurrences at AGN.

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TWX from Northrop Corporation dtd 9/20/63. AEC approval of the installation and use of an in-core irradiation tube is requested.

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University of Illinois ltr dtd 9/17/63 submits proposal to study the kinetics of a sub-critical assembly which is pulsed by the University of Illinois TRIGA reactor. To do this, a natural uranium lattice would be placed in the TRIGA bulk shielding tank. Neutrons from the reactor enter this tank through a graphite column; thus the sub-critical assembly would be driven by the TRIGA in both steady-state and pulsed operation.

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University of Arizona - Issuance of Amendment No. 5 to License; Hazards Analysis; Notice to Office of the Federal Register.

The amendment authorizes the licensee to perform experiments using high temperature test assemblies in its TRIGA Mark I reactor.

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B&W Ltr dtd 8/30/63 and BAW-1284, Ca-7. The B&W Company request a modification to the subject license which will permit the use of a different type of poison in the buffer region of the Lynchburg Source Reactor.

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U. S. Naval Postgraduate School ltr dtd 9/4/63 with enclosures.

The licensee submits additional descriptive material in support of a request for a Part 20 exemption for a high radiation area on the roof of the reactor building.

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General Electric Company (MSCA Facility) - Order of Extension of Completion Date and Notice of Extension of Completion Date dtd 9/30/63.

DL&R has issued an order extending to November 15, 1963, the latest completion date for construction of the Mixed Spectrum Critical Assembly to be located in Building 105 of General Electric Company's Vallecitos Atomic Laboratory.

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Northrop Corporation ltr dtd 9/24/63 and TWX dtd 9/28/63.

AEC approval of the installation and use of a vertical access tube into the core for irradiation of small capacitors and connecting cables is requested.

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Union Carbide Corporation - Amendment No. 5 to Facility License and Federal Register Notice.

The amendment incorporates into the license authorization to possess and use a 50-curie antimony-beryllium reactor startup source and a 300-microcurie strontium-90 nuclear instrument calibration source. The reactor startup source will be leak tested by analysis of pool water.

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Manhattan College Corporation - Order of Extension of Completion Date & Notice of Extension dtd 9/28/63.

DL&R has issued an order extending to 1/31/64 the latest completion date for the construction of a 0.1 watt (thermal), tank-type nuclear reactor on the campus of Manhattan College in New York City.

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United Nuclear Corporation - Proposed Issuance of Construction Permits; Federal Register Notice; Hazards Analysis.

DL&R proposes to issue to United Nuclear Corporation two construction permits which would authorize the construction of (1) a Split Bed Critical Assembly and (2) a Shield Mock-Up Reactor (SMR). Both facilities are to be located in the same new extension to an existing Critical Facilities building at the Corporation's site at Pawling, New York. The reactors are to be located in separate cells, and each would be operated independent of the other from its own control room.

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Worcester Polytechnic Institute - Application dtd 9/3/63 for amendment of Facility License.

The licensee requests a license amendment to reflect a reduction in safety rod scram delay time from 200 to 100 milliseconds. This was made possible by the addition of stainless steel studs which provide an air gap between the magnet face and the armature plate.

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Georgia Institute of Technology ltr dtd 8/30/63 requests approval of plans to conduct hydraulic experiments in the Georgia Tech Research Reactor prior to initial criticality. The tests will use the actual reactor fuel elements, but light instead of heavy water.

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Union Carbide Nuclear Company ltr dtd 9/9/63 reporting results of control rod inspection.

Inspection of the $B_4C$ control rods was conducted after 720 MWD reactor operation. All rods entered the test gage with no evidence of binding.

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General Electric Company ltr dtd 9/20/63 with Amendment No. 7 to License Application.

This amendment requests an extension of the final completion date for construction of the Mixed Spectrum Critical Assembly to 11/15/63.

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Texas A&M University ltr dtd 9/6/63 provides information concerning recent organizational changes at the Nuclear Science Center.

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U. S. Naval Postgraduate School - Tech. Spec. Change No. 1 to License & AEC Hazards Analysis.

DL&R has authorized the licensee to develop and install an automatic power-level control system for its Model AGN-201 reactor.

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Northrop Corporation - Issuance of Change No. 1 to License and AEC Hazards Analysis.

DL&R has approved the installation of an in-core irradiation tube in this TRIGA reactor.

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Union Carbide Corporation ltr dtd 9/24/63 amends a previous application concerning presence of a senior operator at the Union Carbide Nuclear Division Research Reactor Facility.

The licensee now proposes to have a senior operator present during recovery from unscheduled shutdowns or significant power reductions, with the exception only of shutdowns caused by failure of the electrical power supply from the local utility company.

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Western New York Nuclear Research Center, Inc. - Amendment No. 6 and Federal Register Notice.

The amendment, as requested by the licensee, incorporates into the license authorization to possess and use a 30 curie antimony-beryllium source and also provides that the licensee may possess, but not separate, such byproduct material as may be incidentally produced by operation of the reactor. The amendment also adds a condition requiring the licensee to leak test the antimony-beryllium source at intervals not to exceed 12 months.

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Rhode Island & Providence Plantations AEC - Issuance of Amendment No.1 to Construction Permit; Federal Register Notice; and Hazards Analysis.

The amendment authorizes (1) an increase in the design stack height of the reactor from 65 to 115 feet above the surrounding terrain, (2) the inclusion of an auxiliary power supply to insure the availability of a cleanup system in the event of a power failure, (3) a decrease in the differential pressure which the cleanup system would be required to be able to maintain between the building atmosphere and the outside air from 1.5 inches of water to 0.5 inch of water, (4) an increase in the flow rate of air through the cleanup system to the atmosphere from 7 cfm to 3000 cfm and (5) a modification to the reactor plenum.

The DL&R hazards analysis points out that wind tunnel tests employing models of buildings similar to the reactor building indicate that a wind velocity of 30 mph would create low pressures on the lee side of the building such that the pressure outside might be less than the pressure of -0.5 psig inside the building. In this event, gases inside the building might diffuse out through the building walls and penetrations.

The staff concludes, however, that a core meltdown for proposed 1 MW operation is incredible, hence the proposed building confinement is acceptable.

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North Carolina State Collete ltr dtd 10/2/63 amends application for renewal of License No. R-63 to specify a 5-year renewal period.

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University of Florida ltr dtd 10/4/63 provides additional information in support of a requested increase in power of the UFTR from 10 to 100 Kw.

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Industrial Reactor Laboratories, Inc., & Trustees of Columbia University in the City of New York - Amendment No. 13 to License; Federal Register Notice; AEC Hazards Analysis.

The amendment authorizes the licensee (1) to operate the reactor at power levels in excess of 5 Mw, but not in excess of 5-1/2 Mw(t) for periods of time not to exceed 5 minutes in any one week, for the sole purpose of performing routine dynamic tests of the high flux scram feature, and (2) to utilize, within the reactor, core components in which experiment capsules may be placed for irradiation, in accordance with prescribed procedures and limitations.

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U. S. Naval Postgraduate School - Tech. Spec. Change No. 2 to Facility License; AEC Hazards Analysis.

As proposed by the licensee, DL&R has authorized the change in location of the Ra-Be startup source for this AGN-201 reactor during certain transient experiments.

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Northrop Corporation ltr dtd 10/4/63 amends request for AEC authorization of in-core irradiation tube.

This letter rewords the "object" of the request and supplies radiation streaming data.

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Pennsylvania State University ltr dtd 9/20/63 requests a license amendment to permit a slight relocation of the security fence surrounding its Nuclear Reactor Facility (on campus).

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Martin Company ltr dtd 10/7/63 requests extension of License CX-18 until 9/30/64.

Extended contractual discussions have delayed completion of the Liquid Fluidized Bed Reactor program.

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CATEGORY C

Aerojet-General Nucleonics ltr dtd 10/8/63 requests amendment to License R-10 to incorporate existing scram circuitry previously submitted.

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Columbia University - Industrial Reactor Laboratories ltr dtd 9/16/63 supplies additional information requested by DL&R concerning the proposed Advanced Pressure Tube Reactor Critical Experiment.

Among other things, the speed of the rod drive mechanisms has been reduced from 5.4 to 3.3 in/min.

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General Atomic ltr, dtd 9/27/63, with enclosures, supplements an earlier application for a license for critical experiments using a Metal Assembly.

The applicant has now submitted a report on proposed administrative procedures, site data, qualifications of responsible personnel, and corrections to the earlier Hazards Report, GA-4488.

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CATEGORY D

BNL 800 (T-307)

A report "Determination of Uranium at Microgram Levels by Derivative Polarography" dtd May 1963.

The incremental method of derivative polarography has been applied to the low-level analysis of uranium in acetate-buffered EDTA electrolyte. Optimum conditions are shown for the rapid determination of $2 \times 10^{-6}$ M uranium in the presence of 0.01 M bismuth and of $5 \times 10^{-6}$ M uranium in the presence of 0.01 M molybdenum, with a precision $\leq 2\%$ and without prior separation. Uranium in amounts $\leq 10 \mu g$ may be detected in a $10^4$-fold excess of bismuth. The method has been successfully used for the analysis of a bismuth alloy containing 0.1% uranium. (Cpys to JCG, KRO, DAR, & DD - 10/4/63; HJCK, DBH, HWN - 10/21/63).

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SAN 1006
TID 4500 (18th Ed)

A report "Acoustical Measurement of Incipient Boiling" dtd September 1963 by F. H. Misch.

A method for the experimental determination of the time to inception of subcooled nucleate boiling has been developed. It consists of measuring the time between the initiation of an electrical heating step in a thin foil ( or wire) and the acoustical disturbance caused by boiling on the heated surface. (Cpys to HJCK, JCG, HWN & TJT - 10/15/63).

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| Report by Donald W. Pritchard, Johns Hopkins Univ. | "Disposal of Radioactive Wastes in the Ocean" presented at the Symposium on Waste Disposal, Fifth Annual Meeting of the Health Physics Society, Boston, Mass., June 1960.<br><br>An unavoidable consequence of the utilization of atomic energy for power production and other peaceful purposes is the introduction of some radioactive wastes into man's environment. The sea, as one segment of man's environment, will, both purposely and accidentally, receive some of these materials. It is the responsibility of all concerned to assure that such introduction will not endanger man or man's use of the resources of the sea. The purpose of this presentation is to outline the oceanographic problems associated with the peaceful uses of nuclear energy, to indicate what is now known concerning the various pertinent physical, chemical, biological and geological phenomena in the sea which determine the safe capacity of any segment of the sea to receive nuclear materials, and to present a preliminary evaluation of the capacity of the ocean to receive nuclear materials safely. (Distributed at Research Subcommittee Meeting 10/16/63 to WKE, JCG, FAG, HJCK, DAR, LS & RCS). |
| Report by J. Trueman Thompson, Johns Hopkins Univ. | "The Transportation of Highly Radioactive Materials" prepared for the Research Subcommittee of the ACRS, October 1963.<br><br>This statement describes the current status of research and development sponsored by the Division of Reactor Development in the transportation of highly radioactive materials. Particular emphasis is placed on shipping containers of the type currently used to transport high level wastes, useful radiation sources, spent fuel elements, etc. (Distributed at Research Subcte. Mtng. 10/16/63 to WKE, JCG, FAG, HJCK, DAR, LS, RCS). |
| Harvard Report by L. Silverman & R.A. Gussman | "Air Cleaning Studies-Diffusion Board Project - Summary Progress Report, August 1, 1962 - June 30, 1963".<br><br>In the diffusion board concept, a porous building structure would be erected over the reactor or be included in adjacent reactor building walls in the manner of the present steel containment vessel. It permits steam and radioactive gases and particulates to be decontaminated during passage through the board and yet relieves the stored energy without pressure buildup. (Distributed at Research Subcte. Mtng. 10/16/63 to WKE, JCG, FAG, HJCK, DAR, & RCS.) |

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| IDO-16905 | "Nuclear Start-Up of the SPERT IV Reactor" dtd July 1963. |
| | The Spert IV reactor facility is a large pool-type facility designed for the study of the kinetic behavior of a wide variety of reactor types. The first portion of the Spert IV experimental program will be a study of reactor instability as exhibited by highly enriched uranium-aluminum, water-moderated and reflected cores. The first core to be used in this program is a close-packed array of Spert type D 12-plate assemblies. Initial criticality was achieved with this core on July 24, 1962. The initial critical loading contained 21 fuel assemblies which was equivalent to 3.08 kg of U-235. The operational core, a 5 x 5 array of fuel assemblies containing 3.75 kg of U-235, has a maximum cold excess reactivity of 5.3$. |
| | This report describes the initial critical experiment, the operational core loading, and measurements of various static characteristics (void and temperature coefficients, control rod worth, neutron flux distributions, etc) which were made prior to the initiation of the kinetics testing program with this core. Also presented is a brief description of the Spert IV facility. (To HJCK, JCG, HWN, TJT, DBH - 10/22/63). |
| NAA-SR-7909 | "Boiling Studies for Sodium Reactor Safety - Part I" (Experimental Apparatus and Results of Initial Tests and Analysis). |
| | An experimental and analytical research program is described which is designed to meet certain specific needs for data and methods required to make improved predictions of transient voids, burnout, flow, and fuel temperature during extreme accidents in sodium-cooled reactors. Part I of the program has been completed. It includes design, fabrication, and installation of experimental apparatus and preliminary experiments and theoretical analysis. (To DBH, RCS, HJCK, HWN, TJT, DAR - 10/22/63). |
| ANL-6780 | Reactor Development Program Progress Report - August 1963. |
| | Monthly progress report on BORAX-V, liquid metal-cooled reactors, general reactor technology, advanced systems R&D, and nuclear safety. (Cpys to DBH, HJCK, HWN, DAR, & TJT - 10/22/63). |

SUPPLEMENT #45 CATEGORY D

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| UNC-5054 | A United Nuclear Corporation report "Review and Evaluation of Temperature-Induced Feedback Mechanisms in Fast Power Reactors" dtd April 1, 1963.<br><br>The purpose of this work was to investigate the temperature-induced feedback mechanisms which affect the behavior of fast power reactors. It was also to provide a review of methods to predict the reactivity changes caused by these mechanisms.<br><br>The results and findings of an extensive literature review and evaluation are presented, following a reasonably comprehensive development starting from the fundamental relations which describe the dynamic behavior of fast power reactors. (Cpys. to DBH, HJCK, JCG, HWN, & TJT - 10/22/63). |
| ORNL-NSIC-1 | "Effect of Particle Agglomeration on the Penetration of Filters Utilized with Double Containment Systems" dtd 9/25/63.<br><br>This is the first report prepared under the auspices of the Nuclear Safety Information Center.<br><br>The penetration of filter systems in dual containment systems by fission products adsorbed on particulate matter was studied theoretically. For one reactor system treated, less than $10^{-5}$ of the iodine available would be released. It was shown that conditions existing during a reactor accident could not be as pessimistic as those of this model. |
| Report by H. A. Knapp | "Iodine-131 in Fresh Milk and Human Thyroids Following a Single Deposition of Nuclear Test Fallout" dtd 6/1/63.<br><br>The object of this report is to make an estimate of the relation between a single deposition on pasturage of fallout from a nuclear test and the subsequent levels of I-131 in the fresh milk of cows grazing on the pasture, and to use this relation to estimate the I-131 thyroid burdens and radiation doses of persons who daily consume such fresh milk. |
| Committee Review of above report | Committee Review of Report on "Iodine $^{131}$ in Fresh Milk and Human Thyroids Following a Single Deposition of Nuclear Test Fallout".<br><br>The reactions of the individual committee members to the general nature and utility of the report were not in accord, ranging all the way from favorable to |

(Cont'd next page)

Committee Review of Iodine$^{131}$ Rpt. (Cont'd)

completely negative. The basic point at issue is the reliability for this purpose of the gamma ray field measurements from which the author has attempted to derive a quantitative relationship between the extrapolated gamma ray dose rate at 24 hours post detonation and the maximum $I^{131}$ concentration in the milk of cows grazing in the fallout area.

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WCAP-2380

"A Summary of the Design, Fabrication and Quality Control of Large Reactor Pressure Vessels" dtd 9/16/63.

The information contained in this report was compiled to supplement an oral presentation to the DL&R staff on the subject of reactor vessel design. It is expected that this presentation, covering design and manufacture, will establish a basis for acceptance of the general methods and criteria for all vessels, and that detailed design and fabrication reports will not be required for future vessels.

The numerical examples in this presentation are taken from the San Onofre design, however, the methods are generally applicable to the other vessels under consideration. (Distributed at DL&R-Westinghouse mtng. to DAR & RCS - 9/24/63).

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IDO-16885

"Fuel Plate Experience During the Spert I Destructive Test Series with an Aluminum-Clad, Plate-Type Core" dtd July 1963.

This report presents a description of the fuel plate damage which occurred during the Spert I Destructive Test Program performed in 1962, using an aluminum-clad, plate-type fuel. Deformation of fuel plates was observed for power excursions having initial asymptotic reactor periods shorter than 9 msec and melting of plates was observed for tests with periods less than 5 msec. Damage due to pressure pulses was not observed on longer-period tests but was observed during the 3.2-msec-period test in which 35% of the core was melted and violent disassembly of the core occurred. This report includes the results of metallurgical examinations of fuel samples and information on the extent of damage and melting experienced. The fuel plate damage that occurred during these tests is compared with fuel plate damage experienced in other reactors as a consequence of both planned and unplanned excursions.

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