January 8, 1969

50-286

U. S. ATOMIC ENERGY COMMISSION

DIVISION OF REACTOR LICENSING

SUPPLEMENTAL REPORT TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

Note by the Director of the Division of Reactor Licensing

The attached report has been prepared by the Division of Reactor Licensing for consideration by the ACRS at its January 1969 meeting.

TABLE OF CONTENTS

INTRODUCTION

On December 24, 1968, the Division of Reactor Licensing submitted a report to the Advisory Committee on Reactor Safeguards on Indian Point Nuclear Generating Unit No. 3 for consideration at the January 1969 ACRS meeting. The sections of this report dealing with hydrology and flooding, research and development, and quality assurance were not included in that report because our evaluation of the adequacy of the information submitted in these areas was not complete. Since the transmittal of our report to the Committee, the applicant has submitted the Ninth and Tenth Supplements to the PSAR on January 3 and January 6, 1969, respectively, and we have completed our review. Our evaluation of those areas not included in our December 24 report is summarized in this report. The items are numbered to conform with the numbering system used in our December 24, 1968 report to the Committee.

At the ACRS Subcommittee meeting on December 28, 1968, questions were raised concerning onsite electrical power, missiles from rotating machinery inside containment, and the fuel handling accident. The status of our evaluation of these areas is also summarized in this report.

2.5 Hydrology and Flooding

The Indian Point site is located on the east bank of the Hudson River below Peekskill, New York. River water in the vicinity of the plant is used only for industrial cooling purposes. The nearest community utilizing the river for a public water supply is Poughkeepsie, 30 miles upstream of the site. The Chelsea pumping station, 22 miles upstream, can be used as a supplementary source of water supply for New York City. Radionuclides accidentally released to the river would be moved both upstream and downstream by tidal action and many orders of magnitude dilution would occur. Since contaminants would not reach the Chelsea station or Poughkeepsie for at least three tidal cycles, sufficient time would be available to monitor the concentration of the radionuclides in the river and take remedial action, if required, to prevent contamination of the drinking water.

We have reviewed the hydrologic analysis presented by the applicant and conclude that it is adequate. Our consultant, the U. S. Geological Survey (USGS) concurs in this conclusion. The USGS report has been transmitted to the Committee.

Flooding at the site has been evaluated for (1) the fresh water flood resulting from precipitation runoff, the breaching of five major dams upstream of the site, and ebb tide flow and (2) the storm surge associated with the occurrence of the Probable Maximum Hurricane in the vicinity of the site. These were not assumed to occur simultaneously. The applicant has conservatively estimated the flood level associated with the fresh water flood to be 16 ft above mean sea level (+16 ft MSL) and that associated with the

hurricane storm surge to be +19.3 ft MSL. The design water level proposed by the applicant is +19.3 ft MSL. The applicant's analyses of fresh water flood level have been reviewed by the USGS, and the hurricane storm surge calculations have been reviewed by our consultant, the U. S. Army Coastal Engineering Research Center (CERC). Their reports have been transmitted to the Committee. Both the USGS and CERC have concluded, and we agree, that the proposed design water level of +19.3 ft MSL is an acceptable value.

6.0 RESEARCH AND DEVELOPMENT

Specific areas requiring research and development prior to design completion are summarized below:

6.1 Core Stability and Power Distribution Monitoring

The Westinghouse development program on power distribution monitoring consists of correlations between out-of-core measurements and detailed maps derived from in-core instrumentation in operating reactors. The relationship between the indications of the out-of-core detectors and part-length rod position will be developed during detailed design of the core. This study will consider the effects on core stability of both normal operation of the part-length rods and malpositioning of these rods. As experience with operating reactors is gained, detailed information on the effects of core depletion will be correlated with predictions. Although direct experience with the special control rod groups is lacking, it is expected that thiswill be obtained in Ginna and Indian Point Unit No. 2.

We are receiving assistance from Brookhaven National Laboratory and Savannah River Laboratory to aid us in evaluating the problems associated with the detection and control of xenon redistribution. Brookhaven is calculating the response of external and in-core neutron detectors to various power distribution patterns, including those from xenon spatial oscillations. Results thus far have confirmed that external detectors could provide sufficient information to control simple xenon oscillation patterns. Savannah River will examine conditions leading to various modes of xenon oscillations and resulting power patterns and the adequacy of design

provisions for timely detection and control. These results will be of further assistance to us in assessing the adequacy of techniques for coping with xenon oscillations.

It is not completely clear at this time that the Westinghouse program will be able to demonstrate that sufficient information can be derived from external detectors alone. If the planned R&D program does not produce completely convincing evidence that the out-of-core detection system is sufficient, then we will require installation of permanent in-core detectors to furnish information on power distributions. Xenon-produced power redistributions are not highly localized perturbations. Thus, a relatively small number of properly operating axial strings of internal detectors, appropriately distributed radially and azimuthally, should be sufficient for xenon perturbation detection. A research and development program has been undertaken by Westinghouse to develop fixed in-core detectors suitable for continuous monitoring of the core power distribution. Commercially available detectors will be evaluated to determine linearity, response time, sensitivity and lifetime characteristics. Detectors are presently undergoing tests at Yankee-Rowe. Others will be tested at Saxton, San Onofre, and the Western New York Nuclear Research Center. The evaluation will be completed in December 1970. The lifetime evaluation program will extend to the end of 1971.

We conclude that the programs proposed are adequate to determine if the operator will have need for in-core instrumentation. Further, based on the work previously performed by other reactor manufacturers, we conclude that

there is reasonable assurance that a system of fixed in-core detectors can be provided, if required, before operation of Unit No. 3.

6.2 Burnable Poison Rods

This program to be conducted by Westinghouse is designed to verify the calculated reactivity worth of the borosilicate glass rods used to eliminate the potential for a positive moderator coefficient early in the life of the first fuel cycle. The program will also evaluate the effect of the rods on power distribution and the mechanical performance of the rods in the reactor environment. Critical experiments have indicated that the standard methods used in core analysis can be used in the design of a core incorporating burnable poison rods. In-core testing of these rods is underway at Saxton and will continue through mid-1969. These rods will be used for the first time in a commercial power reactor at the Ginna station, scheduled to start up in 1969. Experimental results from these units will be compared with analytic predictions.

We conclude that sufficient time exists to permit the experience gained in in-core testing of these rods to be reflected in the final core design of Unit No. 3.

6.3 Rod Burst Program

This program is being conducted by Westinghouse to determine clad deformation characteristics and the extent of flow blockage under simulated loss-of-coolant accident conditions. The program will include an evaluation of the effects of temperature, prior irradiation and material properties on clad strength and ductility. The effects of void volume, pellet-to-clad gap, pellet cracking, heating rate, initial gas pressure, and metal-water

reaction prior to quenching will be studied. The present schedule is stated below:

Test	Completion Date
1. Rod Burst Tests - Unirradiated Clad	Completed
2. Rod Burst Tests - Unirradiated Hydrided Clad	Completed
3. Complete Quench Tests	December 1968
4. Rod Burst Tests - Irradiated Clad	July 1969

The unirradiated clad experiments (Test 1) indicate that the geometry of the rupture is consistent. It exhibits a small longitudinal split in the cladding with a length of approximately 1/2-inch maximum and a width of 1/32- to 3/16-inch. This rupture would result in a flow area blockage of 10-15% for a single rod. These data also indicate that the experimental burst pressure versus clad temperature curve is 50 to 200% higher than the design curve used in the rod burst analyses. Analytical studies have been completed by Westinghouse assuming that each of the fuel rods forming the hot channel burst at the same axial location to produce the maximum flow blockage possible with an average blockage of 12.5% per rod. The local mass flow rate was reduced to 40% of the nominal value. This reduces the heat transfer coefficient downstream of the rupture and extends the time at which the clad thermal temperature transient reverses. The result is an increase of peak clad temperature of 50°F and a negligible increase in metal-water reaction.

We conclude that the program proposed, in conjunction with the FLECHT program in which Westinghouse is participating, will provide adequate information to establish ECCS effectiveness over a range of parameters

representative of the design basis accident.

Upon completion of our evaluation of program results, a criterion for maximum clad temperature will be established.

6.4 Containment Spray

The additional research and development effort on the containment spray system to be performed by Westinghouse consists basically of data analyses and comparison of calculational models with existing experiments. Little new data will be generated. The program is described below.

1. Droplet coalescence - A theoretical model will be developed which will assume that collisions between spray droplets result in coalescence. The effect on iodine removal will be assessed. The model will be applied to NSPP and CSE experiments and compared with the experimental results. This portion of the program will be completed in the third quarter of 1969.

2. Liquid Phase Mass Transfer Resistance - Liquid film mass transfer resistance will be included in the Westinghouse analytical model. Mass transfer coefficients and partition factors derived from the literature will be applied to determine the effect of liquid film resistance on iodine removal. The model will be applied to NSPP and CSE tests and compared with experimental results to determine if liquid film resistance was significant in the experiments. This program is scheduled for completion in the third quarter of 1969.

3. Materials Compatibility - Tests on the corrosion of major construction materials by the spray solution have been performed. Correlation and documentation of the results of these tests is underway. This

study will investigate the temperature dependency of the corrosion rate over the range of accident conditions. Additional testing is being conducted to investigate corrosion in stressed and welded specimens and the the effect of the spray solution on lubricants, sealants, and insulation. The status of information relative to the use of spray additives will be reported in the first quarter of 1969.

4. Hydrogen Generation - The corrosion data obtained provide a basis for predicting hydrogen generation from metal corrosion. In addition, Westinghouse is presently conducting a program on the effects of flow, temperature, and chemical additives in the radiolytic decomposition of water. Analysis of the data is expected to be completed by the end of 1968.

We conclude that the research and development program on the containment spray system is adequate and will generate sufficient data to analyze the detailed design of the spray system. If the applicant desires to receive additional credit for the removal of inorganic iodine by the sprays, however, the program proposed must be augmented to include (1) data on nozzle performance to indicate spray pattern distribution as a function of distance from the nozzle and (2) graduated scaleup experiments to provide greater accuracy and reliability of iodine removal data.

6.5 Organic Iodine Removal by Charcoal Filters

As indicated in Section 4.2.3.2 of our report to the Committee on Indian Point Unit No. 3, the applicant has proposed an impregnated activated charcoal filter system to be used to remove organic iodides from the

containment atmosphere. As described in the Ninth Supplement to the PSAR, they intend to conduct further R&D in the area of organic iodide removal by impregnated charcoal at high relative humidity, and to reserve space in the containment for installation of dehumidification equipment if the R&D program indicates it is necessary. In order to reduce the relative humidity of the air to below 90%, it is necessary to heat the air a few degrees Fahrenheit. This will require that a heater be installed in each filtration unit having a capacity of less than 100 kw. Such a heater can be designed using standard engineering techniques. Details of the proposed program are discussed below.

Arrangements are being made by Westinghouse to have tests conducted at the Oak Ridge National Laboratory (ORNL) to supplement existing data. These tests are designed to simulate the charcoal beds proposed for Indian Point Unit No. 3 better than the previous ORNL tests. The carbon test bed will be 3 inches in diameter, 2 inches deep, and will be contained between punched plate retainers. It will be oriented in such a manner that tests can be performed with either upflow or downflow through the bed. Tests will be conducted with a methyl iodide concentration of 6 mg/m^3 which simulates the concentration expected in the containment following a loss-of-coolant accident. Most of the previous ORNL tests were performed using a 1-inch diameter carbon bed, 2 inches deep, contained by a wire mesh screen and oriented 35° from the vertical. Most of these tests were run with a methyl iodide concentration of 80 mg/m^3 in the air stream.

The following tests are proposed as a part of this additional R&D effort:

(1) Six tests with the air stream at relative humidities between 90 and 100% and flow downward through the bed.

(2) Two tests under conditions similar to those in (1) above with flow upward through the bed.

(3) Three tests with the bed initially flooded, then purged of excess water by the flow of air at 100% relative humidity, with downflow through the bed.

(4) Three tests under the conditions of (3) above with upflow through the bed.

(5) Two comparison tests using a wire mesh screen to retain the carbon bed in place of the punched metal plate.

(6) Two comparison tests using a methyl iodide concentration of 80 mg/m^3.

Results are expected during the third quarter of 1969.

Based on the information submitted in Seventh and Ninth Supplements to the PSAR and on discussions with R. E. Adams of ORNL, we conclude that the tests proposed will provide sufficient evidence to define the removal efficiency of the charcoal as a function of relative humidity or of adsorbed water on the bed.

The scale-up required to apply these laboratory tests to the commercial filter units must consider the possibility of localized condensation in the bed resulting from the temperature gradients in the time-dependent thermal distribution within the bed. The thermal distribution will vary with time since the bed is initially cold and heat losses may occur from the exterior surfaces of the filtration units when contacted by the containment spray. The determination of localized condensation can be accomplished using standard engineering techniques (three-dimensional time varying heat conduction calculations with appropriate boundary conditions). Once determined, the effect of condensation on flow blockage and the resulting higher velocity and lower residence time in the bed

can be ascertained. Therefore, we conclude that the data generated by the proposed R&D program, applied appropriately, can be used to determine the organic iodide removal characteristics of the commercial air filtration units proposed.

Several alternatives exist in the design of the air handling and filtration unit depending on the results of the R&D program. These are summarized below:

(1) If it can be shown that the filter efficiency with 100% humid air provides the required 5% per pass to reduce the dose received by an individual at the outer boundary of the low population zone for the duration of the accident to 300 rem plus sufficient margin, considering the potential for condensation in the bed due to thermal gradients, the system can be operated as proposed.

(2) If the results show that the effect of 100% relative humidity with no waterlogging reduces removal efficiency to an unacceptable level, dehumidification equipment can be installed.

(3) If it is shown that the removal efficiency is acceptable with 100% relative humidity air but the bed cannot recover from flooding, the filter can be isolated in a tight enclosure during the first portion of the post-accident period.

(4) If it is shown that the direction of flow is a significant factor in improving either the removal efficiency of the bed with air at 100% relative humidity or the ability to recover a flooded bed, the oreintation of the bed can be modified to achieve the desired flow direction.

Based on the above, we conclude that the R&D program proposed is timely and will generate sufficient information to enable design of an air handling and filtration system having the required capability for removing organic iodides from the containment atmosphere.

Westinghouse intends to accomplish this program by financing the ORNL effort through RDT. Westinghouse has had informal discussions with RDT; however, at this time no formal agreement has been consummated. We will inform the Committee of the status of the formal agreement.

Con Ed has informed us orally that the company views the submittal describing the proposed R&D program as a commitment to accomplish the program even if the work cannot be done at ORNL.

6.6 Failed Fuel Monitor

The R&D proposed for the development of instrumentation to promptly detect failed fuel is discussed in Section 3.7 of our report to the Committee.

6.7 Thermal Shock

As discussed in Section 4.4.3, the applicant's analysis of thermal shock following operation of the emergency core cooling program is essentially complete. However, the results are sensitive to both the fracture mechanics properties of heavy section steel and the heat transfer coefficients assumed. The heavy section steel technology program at Oak Ridge National Laboratory will provide information on material properties. It is scheduled for completion by 1973. Westinghouse is making efforts to obtain the effects of temperature and irradiation on fracture toughness. They

participate in a Euratom-funded program to obtain this information.

We conclude that adequate information on the material properties will be available before the vessel experiences the several years of irradiation required to embrittle the steel to the point where failure is considered credible.

6.8 Other Research and Development Programs

Other areas of research and development conducted by Westinghouse are outlined below:

1. Saxton Loose Lattice Irradiation Program to determine fuel performance of standard fuel assemblies at high linear heat generation rate and high burnup. Completion is scheduled for the last half of 1971.
2. Zorita Irradiation Program to determine performance of standard fuel assemblies at high linear heat generation rate and high burnup. Completion is scheduled in April 1973.
3. ESADA DNB Program to experimentally determine the effect non-uniform rod axial heat flux distributions on DNB. Testing will be completed by September 1969.
4. Loss-of-Coolant Analysis Program to incorporate more realistic heat transfer models into the computer codes used to evaluate the consequences of loss of coolant accidents. This program was scheduled for completion on October 1968. We are awaiting a report.
5. FLECHT (Full Length Emergency Cooling Heat Transfer Test) to experimentally determine the thermal behavior of fuel rods during the simulated core recovery period following a loss of coolant accident. It is scheduled for completion by February 1970.

6. Flashing Heat Transfer Program to obtain experimental values of the heat transfer coefficients during blowdown, when uncovered, and during reflooding. Completion was scheduled for October 1968. A report of the final results is expected shortly.

7. Blowdown Force Evaluation Program to determine the forces on core internals during the blowdown. BLODWN-1 has been developed to analyze the pressure velocity and force transients during the subcooled portion of the blowdown. The program is being extended to consider two-phase blowdown. This is scheduled for completion in January 1969.

6.9 Conclusions

Based on our review of the research and development programs proposed, we conclude that these programs are timely and are reasonably designed to accomplish their respective development objectives, will provide adequate information on which to base analyses of the design and performance, and should lead to acceptable designs for the respective systems.

As background regarding the Westinghouse sponsored R&D, we have sent to the Committee, a report titled "Westinghouse Electric Corporation Development Programs." This report was submitted to the Atomic Safety and Licensing Board in support of the Public Service Electric and Gas Company's Salem application. We have also sent to the Committee, as additional background information on R&D programs, reports entitled "Iodine Removal by Sprays" dated October 28, 1968, and "Staff Comments on Zion R&D". Both of these reports were submitted to the Atomic Safety and Licensing Board during the Zion hearing.

8.0 QUALITY ASSURANCE

The Regulatory staff is in the process of developing criteria for guidance in the area of quality assurance. At the present time these criteria are still evolving. For our evaluation of the quality assurance program for the Indian Point No. 3 facility we used the guidelines established for our recently completed review of the Zion application. As expected, the manner and extent to which the quality assurance programs for these two differently owned and designed facilities met with our evaluation guidelines differed.

Indian Point No. 3 is a turnkey project. The applicant (Consolidated Edison) gave to Westinghouse the prime responsibility for assuring adequacy in all design and construction activities. The principal subcontractor, United Engineers and Constructors (UE&C), will prepare all construction specifications and manage all construction work. Most of the quality assurance activities will be carried out by Westinghouse and UE&C. Each of these organizations has a quality assurance organization and each will have a separate quality assurance program for the Indian Point No. 3 project. Consolidated Edison has stated that they will also have a quality assurance organization for this project. The applicant's quality assurance program will include monitoring of the Westinghouse & UE&C efforts. Most of this surveillance will be performed directly for the applicant by the U. S. Testing Company, but some will also be conducted by Consolidated Edison's own engineers.

The overall quality assurance program is described in the application, and in particular in Supplements 1 and 5 to the PSAR. On the basis of this information and recent discussions, we have determined that the applicant's quality assurance program is in general accord with our guidelines. In some areas the program is excellent and exceeds other programs we have reviewed. These areas include (1) the steps being taken to assure close association and interchange of information at all levels between respective functional groups, including those associated with the applicant and all subcontractors, (2) the applicant's decision to have the U. S. Testing Company report directly to the applicant and perform surveillance according to a preliminary, yet carefully delineated USTC surveillance plan, (3) the intent of the applicant to assure that independent checking of designs at important interfaces will be carried out between UE&C and Westinghouse, and between these and other important organizations, and (4) the applicant's intent to use his own engineers in a number of quality assurance activities to supplement those being performed for him by USTC.

There are, however, some areas of the quality assurance program about which we remain concerned and which must yet be resolved. These areas are:

(1) A clearer definition is required of the applicant's exact organization and role in this project, including a detailed discussion of the applicant's internal quality assurance organization, and his other line and staff organizations.

In addition a list of titles of Quality Assurance Procedures and Quality Control Instructions to be used by the applicant to prescribe the QA activities, procedures and efforts to be undertaken and the organizations to be involved is required.

(2) A clearer definition of the organization and programs to be used by the U. S. Testing Corporation is required. A determination must be made as to whether USTC is playing a larger role in quality assurance efforts than it normally performs. A list of the qualification requirements for the USTC quality assurance personnel is also required. In addition a list of titles of Quality Assurance Procedures and Quality Control Instructions to be used by USTC is required.

(3) Lists of titles of Quality Assurance Procedures and Quality Control Instruction to be used by Westinghouse and UE&C is required.

(4) The applicant should indicate more clearly his intent to require Westinghouse and UE&C to provide a specific plan that will assure independent review.

(5) The applicant should provide further assurance that he will have and use a planned systematic procedure for audits, designed to assure independence of inspectors and QA organizations from those responsible for meeting construction or manufacturing budgets and scheduled deadlines. A clearer definition of the manner in which the applicant and Westinghouse will monitor UE&C is required.

We are continuing our discussions with the applicant in order to resolve our concerns. We have just met with the applicant to discuss these areas and will report the results orally to the Committee. We conclude that all outstanding areas of concern can be resolved with the applicant prior to issuance of the construction permit.

In conclusion we believe that, upon successful resolution of the remaining areas of concern, the applicant's Quality Assurance Program will meet our present Quality Assurance guidelines.

ITEMS RAISED BY ACRS SUBCOMMITTEE

1.0 ONSITE ELECTRICAL POWER SYSTEM

As we stated in our previous report to the Committee, we do not consider that there is adequate independence of the diesel generators and their associated buses in the Indian Point 3 design. We do not believe the proposed design meets the intent of Criterion 39.

Our objective is a system with redundant onsite power sources and buses which are independent to the extent that they cannot be connected together with automatically operated circuit breakers. We believe that a system with manual tie breakers can be designed to provide adequate independence. Interlocking of such manual breakers to prevent connecting two redundant operating power sources together would be required. We expect that only control circuit changes to the Indian Point 3 design are required to provide adequate independence and that no additional equipment would be required. The modified circuit should be simpler and actually require less equipment. We have asked the applicant to investigate designs having greater independence, and we will resolve the question prior to installation.

We will further discuss this matter with the applicant immediately prior to the January meeting and will be prepared to report orally to the Committee on the results of that discussion.

2.0 MISSILES FROM ROTATING MACHINERY INSIDE CONTAINMENT

We have initiated a study to assess the adequacy of the design and surveillance provisions to insure the continued integrity of the primary coolant pump flywheels in the Indian Point No. 3 facility during operation. We have also investigated the potential consequences of missiles generated through failure of a flywheel and the design features that could be provided to mitigate these consequences.

Our initial review indicates that the design of the flywheel is conservative; the bursting speed is 3900 rpm versus a design speed of 1500 rpm for a safety factor of 2.6. The applicant has stated that the flywheel will be designed, fabricated, and installed within a rigorous quality assurance program. The flywheel will receive a 100% volumetric ultrasonic inspection at completion of fabrication. An inspection program for the flywheel will be followed during operation and the ultrasonic inspection will be repeated at intervals during the course of plant life. We conclude that the design and fabrication programs for the flywheels are adequate and that an acceptable in-service inspection program can be established prior to operation.

We have investigated the potential for (1) containment liner penetration and (2) primary system penetration, in the event that missiles are generated through failure of a flywheel.

The flywheels are installed horizontally. Review of the PSAR indicates that, with one general exception, sufficient barriers appear to exist along all potential missile paths to prevent a missile from striking the containment

liner. The exception is for a missile traveling in the vertical direction. Vertical motion is possible only through ricochet. The PSAR shows some missile shields over the steam generators but the figures are not detailed enough to show conclusively that a missile moving vertically upward from a pump flywheel will not contact the containment liner.

The flywheel is located about 10 feet from the steam generator and is in the horizontal center plane of the generator shell. In the event a missile were to be generated in the direction of the generator it could potentially breach the shell and release the secondary coolant. Concurrent loss of primary coolant is also possible either through missile induced tube failures or through rupture of a primary piping nozzle as a result of bending moments and shear stresses caused by missile impact. The former occurrence appears more probable than the latter. We have not performed sufficient analyses nor has the applicant provided us with information that would permit us to conclude that this type of potential missile damage need not be considered a basis for design. We believe that, if necessary, adequate missile shielding can be provided to protect the steam generator.

The problem of equipment generated missiles is an industry wide problem and is not limited to the Indian Point No. 3 facility. We intend to continue our general review of this problem for all reactor facilities including review of equipment failures (pump rotors and volutes as well as flywheels) and needed protective features.

3.0 PRELIMINARY REVIEW OF REFUELING ACCIDENT EVALUATIONS

The refueling accident which is generally postulated involves lifting a spent fuel bundle completely out of the core. At some point, either over the core or over the fuel rack in the storage pit, the bundle is inadvertently dropped through the water available during the refueling. For the PWR's, the free fall is stated to be 15-1/2 ft whereas the free fall is presented as 30 ft for the BWR.

Several potential modes of impact have been considered; for example, a direct end impact onto another bundle in the core of the fuel rack may occur, or the bundle may impact on the concrete floor of the fuel pit. For a direct impact, it is categorically stated by the PWR vendors that no fuel pin failure will occur from the longitudinal loading which results. For BWR's on the other hand, GE assumes that the bundle initially impacts at an angle and that the compound loading causes all the pins to rupture. The fuel pin failures are considered to occur after the vertical fall motion is arrested by the end impact and then the bundle rotates about its end to impact broadside against some structural member. The damage is assumed to occur during the rotation.

One other mode, producing the most extensive estimated damage, explored by one applicant, assumes that the bundle falls back into its original slot in the reactor core ricocheting off the sides of the nearby fuel elements in place. Damage is estimated to occur to the falling bundle as well as to the four surrounding bundles. This approach was

presented by Combustion Engineering, Inc. for the Maine Yankee plant (May 1968) but was not included in the Calvert Cliffs analysis by Combustion Engineering, Inc. (December 1968). No explanation was given for the change.

In the GE analysis, the bundle is postulated to impact at a slight angle after which it rotates onto the core top sustaining damage itself as well as damaging portions of other bundles in the core as the fallen bundle rebounds. Most applicants' analyses assume values for the absorption of the kinetic energy of the bundle by structural deformation of suitable numbers of pins. The GE analysis, however, uses the results of some limited measurements which have been made on the energy required to deform an unirradiated tube sufficiently to cause perforation. GE takes credit for the 80-mil-thick Zr channel enclosing the bundle. The PWR fuel bundle designs are not contained within an enclosing channel. Instead, five guide tubes, used for the control rod fingers, form the basic structure of the bundle assembly. Presumably, the energy absorption calculations for the PWR design do not take credit for the energy absorption capability of these guide tubes.

A summary of the results of typical evaluations presented by the applicants for several plants is presented in the accompanying table. It should be noted that in addition to the differences between the basic PWR and BWR fuel bundle designs, other significant differences exist in the various evaluations by the reactor designers. For example, the mechanical behavior after the bundle is assumed to drop varies from a simple line contact with some unspecified structural member impacting on one side, to a bundle falling

back into its core slot and ricocheting against neighboring elements in place in the core. The GE "model" is a variation on the rotation mode wherein the bundle falls in rotation and bounces upon other bundles in the core, thereby extending the damage beyond the fallen bundle itself. The calculational procedures are also not uniform with regard to the assumed energy absorption capability of the fuel pin structures. In the estimation of the radiological consequences of the fuel pin ruptures, the extent of credit taken for plate-out on the plant surfaces, attenuation by the radioactive material path, and the extent of radioactive material released (i.e., from clad-to-pellet void space only or from additional fuel plenum gas) also varies. In this latter area, some alterations of calculated doses have been made by the applicants as a result of discussions with the DRL staff, but uniformity of approach in dose calculations has not been established.

In conclusion, the actual nature of this accident is in doubt. It is necessary to explore the total refueling process to clearly establish the combinations of circumstances which may occur, some of which may pose greater potential hazard than a simple bundle drop. Consideration must be given, for example, to the reliability of the refueling crane, the placement and movement of fuel casks, etc. If the potential refueling accident can lead to significant calculated dose levels, some precautionary measures may be warranted; for example, the installation of ventilating and filtration equipment, or the requirement that refueling be carried out entirely within a closed containment environment.

We plan to accomplish a more complete evaluation of this accident in the context of our construction permit review of the Baltimore Gas and Electric Company Calvert Cliffs plant.

SUMMARY OF REFUELING ACCIDENT EVALUATION

Reactor Plant (Type)	Fuel Description (Bundle & Pin Size)	Accident Site	No. Pins Ruptured
Maine Yankee (CE) (PWR)	14 x 14-176 pins-1250 lbs 0.44" OD, 26 mil Zr clad	Stor. Pit Core-(1) Core-(2)	5 rows-70 5 rows-70 22 rows-
Calvert Cliffs (CE) (PWR)	14x14-176 pins-1250 lbs 0.44" OD, 26 mil Zr clad 5 guide tubes	Stor. Pit Core	1 row-14 none
Ginna (W) (PWR)	14x14-179 pins-1250 lbs 0.422" OD, 24 mil clad	Storage Pit	1 row-14
D. C. Cook (W) (PWR)	15x15-204 pins-1460 lbs 0.422" OD, 24 mil clad	Storage Pit	1 row-15
Indian Pt. No.3 (W) (PWR)	15x15-204 pins-1460 lbs 0.422" OD, 24 mil Zr clad	Storage Pit	1 row-15
Russelville (B&W) (PWR)	15x15-208 pins-1480 lbs 0.42" OD, 26 mil Zr clad	Unspecified Location	56
Hatch (GE) (BWR)	7.7-49 pins-680 lbs 0.563" OD, 32 mil clad	Core	111 (440 upper bound)
Brunswick (GE) (PWR)	7x7-49 pins-680 lbs 0.563" OD, 32 mil clad	Core	111

CONCLUSION

The information presented above completes our summary of the evaluation performed by the Division of Reactor Licensing in our review of Indian Point Nuclear Generating Unit No. 3. The information contained herein in no way changes the conclusion stated in our December 24, 1968 report that the proposed facility can be constructed and operated at the proposed site without undue risk to the health and safety of the public.