Con Edison

REGULATORY DOCKET FILE COPY

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003

August 18, 1978

Re: Indian Point Unit No. 2
Docket No. 50-247

Mr. Boyce H. Grier, Director
Office of Inspection and Enforcement
Region 1
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. Grier,

In accordance with the Technical Specifications of Facility Operating License No. DPR-26, the following confirms telephone notification to Mr. Ted Rebelowski of your office by Mr. John M. Makepeace of Consolidated Edison, this date, of Reportable Occurrence LER-78-022/01T-0. This event is of the type defined in Technical Specification 6.9.1.7.1(i).

On August 15, 1978, discussions with Copes-Vulcan representatives confirmed that a discrepancy existed between the specified and as supplied value of Cv for the new pressurizer power operated relief valves for Indian Point Unit No. 2. These valves (supplied by Copes-Vulcan) were installed during the recently completed refueling outage. The specified value (Cv=50) is the original design value assumed in the analyses for the recently installed Overpressurization Protection System. The as supplied value (Cv=38.5) results in reduced relieving capability for these valves thus permitting operation in a manner less conservative than that previously assumed.

The results of preliminary analyses completed on August 17, 1978, verified that sufficient relieving capacity exists to assure the design limits are not compromised, although previously available margins to these limits are now reduced. We are currently discussing with Copes-Vulcan the schedule for supplying replacement trim for these valves. Further information on our analyses and corrective action will be provided in a follow up report.

Dupe of 7809050016

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Consolidated Edison believes that this report also satisfies the requirements of 10CFR Part 21.

Very truly yours,

William A. Monti

William A. Monti,
Acting Manager
Nuclear Power Generation
Department
Indian Point Station
Buchanan, New York 10511

JMM/mn

cc/ Mr. William G. McDonald, Director (2 copies)
Office of Management Information and Program Control
c/o Distribution Services Branch, DDC, ADM
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

William J. Ca[illegible] Jr.
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

August 4, 1978

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
ATTN: Mr. Victor Stello, Jr., Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

By letter dated July 17, 1978, Con Edison responded to Questions 1 through 8 of your May 17, 1978 information request. The response to Question 9 is provided in the attachment to this letter. This submittal completes Con Edison's response to the aforementioned information request.

Very truly yours,

William J. Cahill, Jr.

William J. Cahill, Jr.
Vice President

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ATTACHMENT

9. Discuss the degree to which your facility complies with the eight (8) regulatory positions delineated in Regulatory Guide 1.13 (Revision 1, December, 1975) regarding Spent Fuel Storage Facility Design Basis.

   A comparison between Regulatory Guide 1.13 (Revision 1, December, 1975) and the Indian Point Unit No. 2 fuel handling building is provided in the following pages.

REGULATORY POSITION 1:

The spent fuel storage facility (including its structures and equipment except as noted in paragraph 6 below) should be designed to Category I seismic requirements.

The Indian Point Unit No. 2 spent fuel storage pit and spent fuel racks are of Seismic Class I design (see pages 9.5-3 and 9.5-6 of Section 9.5 of the Indian Point 2 FSAR and responses to questions B1 and B2 transmitted to the NRC by letter dated May 9, 1975, Mr. Cahill to Mr. Lear). The response to question 1.3 of the FSAR presents a seismic evaluation of the fuel storage building structure.

REGULATORY POSITION 2:

The facility should be designed (a) to keep tornadic winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and (b) to keep missiles generated by tornadic winds from contacting fuel within the pool.

The spent fuel pool tornado protection is discussed in WCAP-7313L "Tornado Induced Water Removal from Spent Fuel Storage", WCAP-7572L "Effect of Torando Missiles on Stored Spent Fuel" and WCAP-7387L "Characteristics of Tornado Generated Missiles".

REGULATORY POSITION 3:

Interlocks should be provided to prevent cranes from passing over stored fuel (or near stored fuel in a manner such that if a crane failed, the load could tip over on stored fuel) when fuel handling is not in progress During fuel handling operations, the interlocks may be bypassed and administrative control used to prevent the crane from carrying loads that are not necessary for fuel handling over the stored fuel or other prohibited areas. The facility should be designed to minimize the need for bypassing such interlocks.

The response to question 9.5 of the Indian Point 2 FSAR describes the use of mechanical stops on the bridge rails. These mechanical stops are used to prevent the crane from passing over stored spent fuel when the spent fuel cask or other similar heavy loads are being handled in the vicinity of the stored spent fuel.

REGULATORY POSITION 4:

A controlled leakage building should enclose the fuel pool. The building should be equipped with an appropriate ventilation and filtration system to limit the potential release of radiactive iodine and other radioactive materials. The building need not be designed to withstand extremely high winds, but leakage should be suitably controlled during refueling operations. The design of the ventilation and filtration system should be based on the assumption that the cladding of all of the fuel rods in one fuel bundle might be breached. The inventory of radioactive materials available for leakage from the building should be based on assumptions given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Safety Guide 25).

The ventilation of the fuel storage building is accomplished by one exhaust fan and two supply fans and heaters. A charcoal filtration system has been installed as part of the exhaust system of the fuel storage building. An exhaust fan in the fan room draws air out of the fuel storage building and exhausts through the plant vent past a process radiation monitor. During refueling operations the normal flow path is blocked by dampers and the air is directed through the charcoal filters before being vented through the stack. The building is maintained at a negative pressure by the exhaust system to insure that leakage is only into and not out of the building.

The response to Question 14.6 of the Indian Point Unit No. 2 FSAR describes an analysis in which all rods in an assembly are breached. In addition, an evaluation performed using the principal assumptions outlined in Regulatory Guide 1.25 also shows that if the cladding of all fuel rods in one fuel assembly is breached, the exposure limits of 10 CFR 100 would not be exceeded.

REGULATORY POSITION 5:

The spent fuel storage facility should have at least one of the following provisions with respect to the handling of heavy loads, including the refueling cask:

a. Cranes capable of carrying heavy loads should be prevented, preferably by design rather than by interlocks, from moving into the vicinity of the pool; or

b. Cranes should be designed to provide single-failure-proof handling of heavy loads, so that a single failure will not result in loss of capability of the crane-handling system to perform its safety function; or

c. The fuel pool should be designed to withstand, without leakage that could uncover the fuel, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted. If this approach is used, design provisions should be made to prevent the crane, when carrying heavy loads, from moving in the vicinity of stored fuel.

The responses to questions 2 and 6, submitted to the NRC by letter dated July 17, 1978, show that spent fuel casks and other similar heavy leads are not moved over stored spent fuel. Mechanical stops (see Regulatory Position 3) are provided to prevent the crane from passing over stored spent fuel when the spent fuel cask or similar heavy loads are being handled in the vicinity of the stored spent fuel. In addition the response to question 9.6 of the Indian Point Unit No. 2 FSAR demonstrates that very little water would be lost from the spent fuel pool if the cask were to drop into the pool.

Drains, permanently connected mechanical or hydraulic systems, and other features that by maloperation or failure could cause loss of coolant that would uncover fuel should not be installed or included in the design. Systems for maintaining water quality and quantity should be designed so that any maloperation or failure of such systems (including failures resulting from the Safe Shutdown Earthquake) will not cause fuel to be uncovered. These systems need not otherwise meet Category I seismic requirements.

Loss of coolant that would uncover the spent fuel is unlikely to occur for the following reasons:

(1) There are no drains on the bottom or side walls of the spent fuel pit.

(2) The suction of the spent fuel pit cooling pump is taken from a point approximately six feet below the surface of the pool.

(3) The spent fuel pit pump discharges into the pool approximately seven feet above the top of the spent fuel assemblies. This discharge line also has a hole drilled in it preventing it from becoming a syphon and partially draining the pit.

(4) The skimmer pump takes suction from and discharges to the surface of the pool.

Thus, the failure of the spent fuel cooling loop and/or clean-up equipment would not result in the uncovering of the spent fuel.

REGULATORY POSITION 7:

Reliable and frequently tested monitoring equipment should be provided to alarm both locally and in a continuously manned location if the water level in the fuel storage pool falls below a predetermined level or if high local-radiation levels are experienced. The high-radioactive-level instrumentation should also actuate the filtration system.

Gamma radiation levels in the fuel storage building are continuously monitored by a local area radiation monitor. This monitor provides alarms both locally and in the control room if the water level in the pool is low or if high local radiation levels are experienced. This instrument is designed to provide automatic actuation of the filtration system.

REGULATORY POSITION 8:

A seismic Category I makeup system should be provided to add coolant to the pool. Appropriate redundancy or a backup system for filling the pool from a reliable source, such as a lake, river, or onsite seismic Category I water-storage facility, should be provided. If a backup system is used, it need not be a permanently installed system. The capacity of the makeup systems should be such that water can be supplied at a rate determined by consideration of the leakage rate that would be expected as the result of damage to the fuel storage pool from the dropping of loads, from earthquakes, or from missiles originating in high winds.*

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*The staff is considering the development of additional guidance concerning protection against missiles that might be generated by plant failures such as turbine failures. For the present, the protection of the fuel pool against such missiles will be evaluated on a case-by-case basis.

The response to Regulatory Position 8 is provided in the following documents:

(1) Response to Regulatory Position 2.

(2) Response to Question 9.6 of the Indian Point Unit No. 2 FSAR.

(3) Response to Question B.1 submitted to the NRC by letter dated May 9, 1975.

(4) Appendix A of the Indian Point Unit No. 2 FSAR.

(5) Section 9.2 of the Indian Point Unit No. 2 FSAR.

(6) Section 9.3 of the Indian Point Unit No. 2 FSAR.

William J. Cahill, Jr.
Vice President

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Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

July 17, 1978

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
ATTN: Mr. Victor Stello, Jr., Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

In response to your letter dated May 17, 1978 answers to questions 1 through 8 are provided in the attachment to this letter. The response to question 9 will be provided to you by August 4, 1978.

Very truly yours,

William J. Cahill, Jr.
Vice President

attach.

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ATTACHMENT

1. Provide a diagram which illustrates the physical relation between the reactor core, the fuel transfer canal, the spent fuel storage pool and the set down, receiving or storage areas for any heavy loads moved on the refueling floor.

   Figures 9.5-1 and 5.1-6 of the Indian Point Unit No. 2 FSAR illustrate the physical relation between the reactor core, the fuel transfer canal and the spent fuel storage pool. Figures 5.1-2 and 5.1-6 of the FSAR show typical laydown areas including the reactor head and upper internals laydown areas. Heavy loads moved on the refueling floor for maintenance reasons do not have predetermined laydown areas.

2. Provide a list of all objects that are required to be moved over the reactor core (during refueling), or the spent fuel storage pool. For each object listed, provide its approximate weight and size, a diagram of the movement path utilized (including carrying height) and the frequency of movement.

In addition to the reactor head (approximately 169 tons including the rig) and the upper internals (approximately 63 tons including the rig) which have to be moved over the reactor each refueling other heavy loads could be moved for maintenance, inservice inspection etc. Examples of such heavy loads include:

1. reactor coolant pump motor:

   weight: 32 tons
   size: 6' round x 12' long (approx.)
   movement: no set path of movement
   frequency of movement: varies

2. inservice inspection tool:

   weight: 5 tons
   size: 14' round x 20' long (approx.)
   movement: no set path of movement
   frequency of movement: 2 times per outage

The only heavy object that would normally be required to be moved over the spent fuel storage pool is a cask. Technical Specification 3.8.A.7 states: "If the spent fuel pit contains spent fuel, the spent fuel cask shall not be moved over any region of the spent fuel pit until the cask handling system has been reviewed by the Nuclear Regulatory Commission and found to be acceptable." Therefore, there are no such heavy objects that will be moved over the spent fuel storage pool without first having the required prior review by NRC.

3. What are the dimensions and weights of the spent fuel casks that are or will be used at your facility?

At the present time Con Edison does not own any spent fuel casks for Indian Point 2 spent fuel. Con Edison does not have any contracts with vendors to supply casks, at this time.

4. Identify any heavy load or cask drop analyses performed to date for your facility. Provide a copy of all such analyses not previously submitted to the NRC staff.

   Excluding fuel assembly drop analyses, which are not considered heavy load analyses, the following is a list of documents where heavy load or cask drop analyses were previously described to the Commission:

   1. Question 9.6 of the Indian Point Unit No. 2 FSAR.

   2. Letter from Mr. W. J. Cahill, Jr., Con Edison to Mr. George Lear, NRC, dated July 23, 1975.

5. Identify any heavy loads that are carried over equipment required for the safe shut down of a plant that is operating at the time the load is moved. Identify what equipment could be affected in the event of a heavy load handling accident (piping, cabling, pumps, etc.) and discuss the feasiblity of such an accident affecting this equipment. Describe the basis for your conclusions.

Heavy loads are not carried directly over equipment required for the safe shutdown of Indian Point 2 when the equipment is operating. All safety systems are located in the primary auxiliary building or below the 95' elevation of the vapor containment building. Movement of heavy loads on the 95' elevation would not endanger equipment used for the safe shutdown of the plant that is operating at the time the load is moved.

6. If heavy loads are required to be carried over the spent fuel stoage pool or fuel transfer canal at your facility, discuss the feasibility of a handling accident which could result in water leakage severe enough to uncover the spent fuel. Describe the basis for you conclusions.

Heavy loads are not carried over the spent fuel pool or fuel transfer canal. See response to question 2.

7. Describe any design features of your facility which affect the potential for a heavy load handling accident involving spent fuel, e.g., utilization of a single failure-proof crane.

The responses to questions 9.5 and 9.6 of the Indian Point 2 FSAR desribe the use of mechanical stops on the bridge rails and conservative design margins used for the cask related handling equipment, respectively.

8. Provide copies of all procedures currently in effect at your facility for the movement of heavy loads over the reactor core during refueling, the spent fuel storage pool, or equipment required for the safe shutdown of a plant that is operating at the time the move occurs.

For the reasons presented in responses to Questions 5,6 and 7 Con Edison does not have specific procedures which address the movement of heavy loads over the reactor core during refueling, the spent fuel pool or the equipment required for the safe shutdown of the unit that is operating at the time the move occurs. Sections of a procedure which pertains to the lifting of the reactor head and internals is provided for your information.

Con Edison

Indian Point Station

Maintenance Procedure

Title: Reactor Disassembly And Reassembly; For Refueling, Inservice Inspection And/Or Extraordinary Maintenance

Identification Number: 2-CM-2.4

2.4	RVI	Reactor	Head & Internals
Number	System	Sub-System	Component

Revision Number: 1

Date: 1/25/78

Page Number 1 of 191

NOTE: THIS ATTACHMENT CONTAINS ONLY THOSE SECTIONS WHICH PERTAIN TO THE LIFTING OF THE REACTOR HEAD AND UPPER INTERNALS.

Prepared By J. Quirk/W. Monti (October 1977)

QA Engineer: William Luraine 1/18/78

Reviewed By: [signature] 1/24/78

J. DeVita Meeting 116,

SNSC Review/Date: N.C. Wallace 1-25-78

NFSC Review/Date: Not Required

Approved By: [signature] 1/25/78

MWR No. 2N5

A. PURPOSE

To provide a detailed procedure for the removal and replacement of the Reactor head, Reactor internals, and associated hardware in preparation for and returning from refueling operations, inservice inspection(s) and/or extraordinary maintenance operations.

NOTE

Before proceeding with work review sections B to D thoroughly.

B. PREREQUISITES

1.0 Appendix B and Appendix D respectively contain consumable material and special tool lists; the responsible General Maintenance Supervisor should assure the availability of this material and these tools well in advance of actual need.

2.0 Prior to beginning the actual work an MWR shall be obtained from the Operations sub-section and the number recorded on the face of this procedure (see SAO-104).

3.0 In order to proceed with work there is a need to obtain Work Permits, Radiation Work Permits and a Primary Pressure Boundary Permit. The acquisition of these permits is the Maintenance Supervisor's responsibility. These permits should be returned to the issuing authority as soon as their need is no longer valid. The General Maintenance Supervisor should keep, as part of this procedure's use, a catalog of open and closed permits. (See SAO-105 and procedure QA-10.)

C. PRECAUTIONS

1.0 RADIOLOGICAL

1.1 At several steps in this procedure the Reactor Coolant System (RCS) will be opened to containment atmosphere. Be aware of possible presence of radiogas, contaminated liquid and particulate contamination. Where surfaces or areas exposed to the RCS are opened, radiation streaming may be experienced. Exposed contaminated surfaces will generate both gamma and beta fields. Beta fields can cause exposure to the skin of the whole body and to the lens of the eye unless these areas are protected. In certain cases high contact field readings will require the individual to wear wrist badges or finger rings. These items will be specified by H.P. and specified on the RWP when required. Observe radiological precautions indicated in this procedure and required by the Radiation Work Permits (RWP).

1.2 Use care in the disposal of any contaminated or radioactive material. All material removed for disposal must be properly wrapped in plastic and approved by Health Physics prior to being removed from the job site.

1.3 Any materials or parts stored for reuse must be carefully wrapped, and marked to identify them to prevent inadvertent disposal.

1.4 All personnel involved in the job shall be instructed in the radiological protection requirements, and be familiar with the radiological rules and guidelines in effect at the plant site.

1.5 All individuals should be thoroughly familiar with their job functions when working in high radiation areas in order that the ALARA concept of collective dose (man-rem) to all personnel and to the individual be kept as low as reasonably achievable.

2.0 SYSTEM INTEGRITY

2.1 Any opening to the RCS must be sealed with plastic and yellow tape when work is not in progress or a person assigned to monitor the opening to prevent foreign material from entering system.

2.2 When it is possible for a tool to fall into the open RCS if inadvertently dropped, it must be secured with a line. The QA Engineer will determine when such precautions are necessary.

2.3 The Operations, Maintenance and QA Engineers will jointly determine the extent of the clean area surrounding an opening in the RCS. When working within this area QA - Procedure #10 must be followed.

3.0 QUALITY ASSURANCE - GENERAL PRECAUTIONS

3.1 Notify Q.C. prior to the start of work.

3.2 Full inventory control and equipment accountability is to be maintained in accordance with QA Procedure #10.

3.3 When working near the open RCS all personnel shall wear protective clothing with wrists, ankles, front fly, and pockets taped closed; eyeglasses will be taped to the head or adequately tied to prevent dropping.

3.4 There shall be no other work undertaken in the area (grinding, machining, welding, etc.) which could endanger the clean area or cause foreign material to enter the primary system.

3.5 Calibrated tools and/or instruments are to be used for all specified measurements.

3.6 Nylon line will be used. Manilla rope is not acceptable in the Class A cleanliness boundary.

4.0 SAFETY

4.1 Observe standard safety precautions. Remember that anti-C garments - which are very necessary - make task efforts more tedious and thus might suggest unsafe short cuts. Do not be convinced without thorough review.

D. GENERAL INSTRUCTIONS

1.0 Maintain the Check Off List (COL), section G throughout this procedure. Steps marked (*) require entry on the COL, any step requiring sign-off or data entry are so marked.

2.0 Use of Travelers

Further amplification of any instruction required in the body of this procedure such as the removal of a stuck stud or alignment bushing will be provided in the form of a traveler ammended to this procedure. These will not be considered a procedure change. Such travelers will be prepared and approved by Maintenance Engineer Sub-Section, and reviewed by Q.A. for hold points prior to use.

3.0 Sequence of Steps

The number sequence of steps indicated in this procedure need not be followed except where specifically indicated in the procedure or as indicated below.

Steps 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, and 21 must be done in order.

4.0 Changes to this procedure not covered by 2 or 3 above, must be in accordance with Maintenance Engineer Sub-Section Administrative Directives (AD-4).

CAUTION

Temporary procedure changes must be reviewed by SNSC within seven days.

5.0 Because there is potential for movement of core components (fuel or RCC's) during head and upper internals removal, constant communication between personnel on 95' elevation containment and the CCR must be maintained as specified in this procedure.

E. REFERENCES - Copies are available from the Indian Point Central Files Controller.

1. Westinghouse Refueling Procedure F-2, from Plant Manual.

2. Combustion Engineering Instruction Manual Number 17765, Reactor Vessel.

3. Stud Tensioner Instruction Book, Westinghouse Order # 54-F-66181-B, Biach Industries Order No. 725.

4. Reactor Vessel Insulation Book, Westinghouse Order # 54-F-70611D-M1C369.

5. Westinghouse Instruction F-7.6, Internals Lifting Rig, PSE/PNJ-FHSTIR.

6. Unit No. 2 System Description No. 17, Fuel and Core Component Handling.

7. SAO-105, Work Permits.

8. QA Procedure No. 10, Cleanliness Control.

9. Stearns-Roger Instruction Book B32920, Fuel Transfer System.

10. Royal Industries, Part Length Control Rod Drive Manual No. 106.

11. Unit No. 2 System Description No. 2, Reactor Vessel and Internals.

12. Unit No. 2 System Description No. 14, Incore Instrumentation.

13. Whiting Corporation, Polar Crane Manual for Crane Serial No. 9548 (UE&C F.P. 9321-1289).

14. HPP 2.1 - Radiation Work Permits.

F. DETAILED PROCEDURE

INTRODUCTION

I. This procedure is composed of twenty-five (25) independent sections and one sign-off section, the Check Off List (COL).

II. Work Permits, Radiation Work Permits and the Primary Pressure Boundary Permit can be sought on the basis of each section. This does not mean that one permit set will suffice for each section but rather you can discuss total permit need for each section independently.

WARNING

III. No water, regardless of purpose, shall be used or sprayed on elevation 69' of the reactor cavity until the Presray Seal is installed and pressurized. If this is not done and water is introduced at elevation 69' you will destroy the Nuclear Instrumentation.

11.0 Reactor Head Removal (Refer to Figures 25, 28 and 29)

11.1 Prerequisites

11.1.1 Insure "O" ring gaskets for head are available. See Section 11 of Appendix B.

11.1.2 P.G.M. crane maintenance personnel available in V.C. when lift is made.

11.1.3 Three (3) precision levels.

11.1.4 Primary Pressure Boundary Permit (see Section E, Reference 8).

11.2 Precautions

*11.2.1 Do not proceed until part length CRD shafts have been unlatched. Verify this fact with the Operations Watch Supervisor.

11.2.2 As head is being lifted constant communication shall be maintained with the CCR Operator in case an RCC comes with the reactor head.

11.3 Procedure Steps

11.3.1 Remove all tools and equipment from pit, do not proceed unless step 10.3.6 has been signed off by the Quality Control Inspector.

*11.3.2 Prepare reactor head pad on 95' elevation to receive the head by placing plastic on floor to contain any water, then place two (2) new reactor vessel closure O-rings around closure head storage pad.

CAUTION

Visually inspect new gaskets. Insure that they are free of dents, scratches, and pealing or flaking of coating.

11.3.3 Before proceeding you must verify the following:

A. From Operations Watch Supervisor - that the Part Length Rods are unlatched. This should have been accomplished shortly after step 3.3.7 was completed.

B. That steps 1.3.1 and 1.3.2 have been completed AND signed off on the Check Off List.

*11.3.4 Attach the reactor head lifting rig to the crane and in turn to the reactor head. On the reactor head flange mark stud hole 12 as "12", stud hole 28 as "28", and stud hole 44 as "44". This will facilitate proper orientation when the reactor head is to be reinstalled. Insure the alignment pins are correspondingly marked and boldly with Nissan low chloride marker.

NOTE

The reactor head and its lifting frame weight 169 tons.

*11.3.5 Notify the CCR Operator you are about to lift the Reactor Head. Lift head approximately 1" (by eye), and check for levelness with the three precision levels installed 120 degrees apart. If not, set head back down and adjust head lifting device - sling assembly. Repeat this operation until head is level to within 0.003" per foot to the flange mating surface.

*11.3.6 Notify CCR that reactor head will now be moved up two feet higher. Lift head two feet (check for levelness during lift), insure reactor head does not bind on alignment pins.

*11.3.7 Continue raising head to about 170" above flange. Maintain communication with CCR. Make continuous visual inspection of full and part length rods to assure rods are not moving up with reactor head.

*11.3.8 Lift reactor head out of pit and place it on the storage pad on 95' elevation.

*11.3.9 Have Operations inflate pres-ray seal if not already done as part of step 10.3.2d. Notify Operations that step 11.3.8 is complete so that full length CRDM's may be unlatched. Hold at this point until these items are completed.

*11.3.10 Operations can now flood the reactor cavity.

WARNING

The next part of this procedure involves movement of the reactor's upper internals. These internals weight about 130,000 pounds and are moved under water. Clearances are very tight and require precision actions. CARE is the key word in your dealing with them. You shall review this next section totally before you act and in particular the drawings associated with this section. You must, to be successful, also review it with your men. There is no recovery from damaged upper internals. C A R E.

12.0 Removal of Upper Internals from Reactor Vessel (See Figures 26, 28, 30, 31, 32, and 33)

12.1 Prerequisites

12.1.1 For safe horizontal movement of the upper internals it is necessary that the water be sufficiently clear, so that the Maintenance Supervisor in charge of lifting the upper internals can clearly see the reactor vessel flange with the water level at the normal refueling height of about 93.5'.

12.1.2 One 50' metal tape measure and brass plum bob.

12.1.3 Verification of completion of Section 1.4 of this procedure.

12.1.4 PGM Crane and Elevator group on hand when lift is started.

12.2 Precautions

12.2.1 This move and that of section F16 of this procedure are the most critical. Work must be done in a careful and precise manner, paying close attention to this procedure. There shall be no rush work in an attempt to meet schedule requirements during this phase.

12.2.2 It is necessary for the upper flange of the internals package to clear water by approximately 2 feet. There will be considerable radiation streaming from this exposed portion. H.P. must be on hand to monitor radiation. Personnel within view of the refueling pool must be limited to only those essential personnel for this operation as determined by the Maintenance Supervisor in charge.

12.2.3 Communication with the CCR shall be maintained during this phase until the package is landed on the storage stand.

12.2.4 When working near or over the refueling canal, all tools or equipment which could fall into the refueling canal must be secured by lanyards.

12.3 Procedure Steps

12.3.1 Refer to Figure 26 for an elevation view of the reactor vessel storage stand and internals transport path. This figure shows reference heights necessary to assure clearance between the internals, storage stand, and reactor vessel.

*12.3.2 Make a final inspection of the upper internals lifting rig and the manipulator bridge crane. Insure that they are free of loose particles of dirt or tools, etc., that could fall into the vessel.

*12.3.3 At least two (2) hours prior to anticipated lifting of the internals lifting rig, turn on the Dillon Load Cell indicator and allow it to warm up.

*12.3.4 Obtain reference dimension of upper internals package in vessel for use as a guide in reassembly as follows:

a) Mark a reference location on the manipulator crane from which to drop the measuring tape and plum bob. Record this position for reassembly.

CAUTION

Attach a lanyard to the tape measure. Insure that plum bob is securely fastened to the tape measure.

b) Have Operations position the manipulator crane over the upper internals such that the plum bob can be positioned on the upper internals. Have manipulator operator record the manipulator coordinates of this position and record them on the COL.

c) Lower the tape measure until the plum bob rests on the upper internals. Record the tape reading on the COL.

d) Retrieve the tape measure. Use care handling since it is wet with radioactive water. Place in plastic bag before removal from manipulator.

e) Move manipulator from core area.

STOP

Do not start unless you intend to remain in Containment until the end of step 12.3.5h. The following work cannot be left in mid-stream.

*12.3.5 Latch the upper internals as follows:

a) Lifting Rig Description (Refer to Figures 30, 31 and 32)

The internals lifting rig is a structural frame device to handle the upper and lower reactor vessel internal packages. The rig consists of a sling assembly, spreader assembly, leg assembly, support ring, protective ring, and a mechanism handling tool.

The rig is suspended from the main crane hook and is remotely attached to the internal packages by the use of a mechanism handling tool. This tool is operated from the platform on the manipulator crane.

For initial rough alignment and guidance when over the reactor vessel and upper internals storage stands, the rig uses three long guide studs. For final fine alignment and guidance, the rig uses the four reactor vessel alignment keys and similar keys on the upper internals storage stand. The brackets at the bottom of the rig which guide on these alignment pins must be manually moved when switching from handling the upper internals to handling the lower internals assembly. The brackets must be bolted in the lower set of holes to handle the upper internals and in the upper set of holes to handle the lower internals.

A protective ring assembly, that protects the reactor vessel mating surface during the refueling operations, is supplied with the rig.

The ring is placed on the reactor vessel after the vessel head is removed and remains in place until the refueling operations are complete and the vessel head is to be re-installed. The mechanism handling tool is used to connect and disconnect the protective ring and the rig. For storage, the ring remains attached to the rig.

*b) Install the Dillon Load Cell sensor assembly to the main crane hook by inserting the pin through the two side plates and the hole provided in the hook. After the attachment is complete, suspend the assembly from the hook and visually inspect each component. Insure that the adapters and Dillon Load Cell sensor are free to pivot about the pin connection. Attach a guide line to each side of the sister hook for the purpose of rotating the hook assembly. At no time shall a twisting moment to rotate the hook be induced through the internals lifting rig.

c) Lower the load cell assembly positioning the adaptar between the two top lugs of the internals lifting rig. Insert the connecting pin by use of the pull rod assembly which is attached to one of the top lugs. When the connecting pin extends through the adaptar to approximately 1/8 inch beyond the other lug, tighten the pull rod assembly lock nut. Move the manipulator crane east to a position beyond the reactor vessel to allow access for removal of the internals package. Insure that the four alignment pin brackets are bolted and pinned in the lower set of holes on the rig.

*d) Slowly raise the internals lifting rig, when it is unseated record the Dillon Load Cell reading on the COL. Then proceed to the reactor vessel area. Using the guide lines that are attached to the sister hook, orient the rig with respect to the reactor vessel guide studs, then lower the assembly until it has seated on the upper internals package. (See Figures 26, 28, 30, 31, and 32.)

e) Locate the manipulator crane platform near the center of the internals lifting rig and disconnect the load cell assembly with the pull rod assembly. Remove the load cell assembly from the immediate area.

*f) Attach the mechanism handling tool to the hoist on the manipulator crane, then proceed to the attachment point. Disconnect the protective ring assembly with the mechanism handling tool, and thread the torque tube assembly into the internal package. This is accomplished by placing the adaptar on the end of the tool over the adaptars on the protective ring and torque tube mechanisms and then rotating the handwheel on the top of the tool. Repeat the procedure for each of the three attachment locations. (Approximately: at 15° handwheel, 8 turns.) During the locating movements of the manipulator crane, (at 145° and 250° handwheel 1-2 turns) insure that the mechanism handling tool and the manipulator mast are clear of obstructions.

g) When the rig-internals attachment is complete and the protective ring assembly is disconnected, locate the manipulator crane platform near the center of the rig and connect the load cell assembly as previously outlined. Remove the manipulator crane from the area insuring that the path to the upper internals storage stand is clear.

*h) Insure that the crane is directly over the center position, then permanently bench mark the crane rail with reference to the cranes position. For future use record the location of this bench mark.

STOP

Do not start unless you intend to remain in Containment until the end of step 12.3.6h. The following work cannot be left in mid-stream.

*12.3.6 Raise and transport as follows:

*a) Using critical lift control, begin taking a strain with the polar crane. Observe the Dillon Load Cell indicator for abnormal variation full load should be about 130,000 pounds when full load is obtained raise about 2 inches and check for any abnormal indication such as binding or shifting. <u>Record the actual full load on the COL</u>.

*b) If no abnormal conditions are found, continue to raise slowly until a point is reached where the levelness of the lift may be determined. At this point stop, insure levelness. Monitor the load cell for any deviations from full load. <u>If at any time the load exceeds the expected load by 10% or more, S T O P</u>. If at any point abnormal indications are found the package is to be lowered to rest in the reactor and the Maintenance Engineer notified. If movement in either direction, up or down seems abnormal, stop and notify the Maintenance Engineer. From the point that the lift is begun until the package is landed the crane must not be left unattended. Observe precaution 12.2.2.

c) When the Maintenance Supervisor has assured himself that no abnormal indications are present he may direct more rapid up crane movement. Raise the upper internals package so that the upper plate is sufficiently above 95' elevation for the guide bushing to clear the water surface. (See Figure 26.)

NOTE

Photographs are available in machinery history file showing position of internals package out of water to meet the requirements of this step.

WARNING

At no time shall a twisting moment to rotate the hook be induced through the internals lifting rig. Use the guide lines.

d) Rotate the load 180 degrees and slowly transport the upper internals package to the storage stand.

e) Align the guide bushings with the alignment pins, and lower to within several inches of the stand. (See Figure 26.)

f) Using critical lift speed, lower onto the stand. Observe the Dillon Load Cell indicator as the load is being relieved. Be alert to any shifting or binding during lowering or resting.

g) Once the internals are seated on the stand disconnect the crane from the internals lifting package. The Dillon Load Cell sensor may remain or be removed.

h) Inform CCR that this phase is complete.

12.3.7 Clean up the area, remove and store all tools; proceed to Section 14, Preparation for Reassembly of Reactor Head and Upper Internals.

WARNING

> The next part of this procedure involves movement of the reactor's upper internals. These internals weigh about 130,000 pounds and are moved under water. Clearances are very tight and require precision actions. CARE is the key word in your dealing with them. You shall review this next section totally before you act and in particular the drawings associated with this section. You must, to be successful, also review it with your men. There is no recovery from damaged upper internals. C A R E.

16.0 Reassembly of Upper Internals in Reactor Vessel (See Figures 26, 28, 30, 31, 32, and 33)

16.1 Prerequisites

16.1.1 For safe horizontal movement of the upper internals, it is necessary that the water be sufficiently clear so that the Maintenance Supervisor in charge of lifting the upper internals can clearly see the reactor vessel flange. With the water level at the normal refueling height (about 93.5').

16.1.2 One 50' metal tape measure and brass plum bob from 12.1.2. (Tape must be calibrated.)

16.1.3 PGM Crane and Elevator group on site when lift is started.

16.2 Precautions

16.2.1 This lift and the lift section 15 of this procedure are the most critical. Work must be done in a careful and precise manner, paying close attention to this procedure. There shall be no rush work in an attempt to meet schedule requirements during this phase.

16.2.2 It is necessary for the upper flange of the internals package to clear the water by approximately 2 feet. There will be considerable radiation streaming from this exposed portion. H.P. must be on hand to monitor radiation. Personnel within view of the refueling pool must be limited to only those essential personnel for this operation as determined by the Maintenance and Watch Supervisors.

16.2.3 When working near or over the refueling canal, all tools or equipment which could fall into the refueling canal must be secured by lanyards.

16.3 Procedure

*16.3.1 Connect the crane to the internals lifting package. The Dillon Load Cell sensor must be reconnected if removed. The Dillon Load Cell indicator requires 2 hours warm up prior to its use.

STOP

> Do not start unless you intend not to leave Containment until the end of step 16.3.7. The following work cannot be left in mid-stream.

16.3.2 Using critical lift speed, raise the package off the stand. Observe the Dillon Load Cell indicator as the load is being increased. The load should be the same as when it was removed, see the COL. Be alert to any shifting or binding during movement.

*16.3.3 Raise sufficiently high to clear the reactor alignment pins. Observe precaution 16.2.2. (See Figure 26.)

CAUTION

At no time shall a twisting moment to rotate the hook be induced through the internals lifting rig. Use the guide lines.

*16.3.4 Rotate package 180°, and carefully transport to position over the reactor vessel.

*16.3.5 Align bushings (lifting rig) to alignment pins (in the Reactor Vessel) and lower package to within six (6) inches of landing. This should be determined using the tape and plum bob over from the manipulator. Use care when lowering to assure against binding or shifting. Constantly monitor the load cell. If a 10% change is observed, S T O P and notify the Maintenance Engineer. The cell should read about 130,000 pounds. (See COL for exact load at removal.)

*16.3.6 When the six (6) inch point is reached use the critical lift control to lower the final six (6) inches. Observe the Dillon Load Cell indicator as the load is seated. (See Figure 26.) Remove all load from the crane.

*16.3.7 When the upper internals package has been seated, have Operations reposition manipulator bridge to coordinate with what was recorded in 12.3.4b. Drop plum bob and tape (tape must be secured by lanyard). Insure that plum bob is secure on the tape. Record height above internals, this must agree with the value recorded in step 12.3.4b, if not, notify the Maintenance Engineer. Record reading on COL. Reel up tape and place in plastic bag. Store until next refueling. Log storage position.

*16.3.8 Using mechanism handling tool, unlatch rig and reconnect protective ring. Remove manipulator from over core.

WARNING

The torque tubes are spring loaded, use caution in unlatching.

*16.3.9 Using critical lift speed raise up upper internals lifting rig, the weight must be the same as that recorded in step 12.3.5d on the COL, and place on the upper internals storage stand. Record weight on the COL.

16.3.10 Disconnect the lifting rig from the crane; remove and store the Dillon Load Cell sensor and indicator outside the VC where the Maintenance Engineer directs.

*16.3.11 Notify Operations that the full length CRDM's can be re-latched and the pit can be drained. The reactor cavity can be cleaned before proceeding to next part of procedure but is not mandatory at this time.

CAUTION

The Pres-ray seal must remain inflated during reactor cavity cleaning.

17.0 Reactor Head Replacement (See Figures 25, 28 and 29)

17.1 Prerequisites

17.1.1 Verification of completion of step 14.2 above.

17.1.2 PGM Crane and Elevator group shall inspect the Polar Crane prior to lift. Crane and Elevator personnel shall be on site until lift is complete.

17.2 Precautions

17.2.1 Until the head is landed, there will be a high radiation field coming from the open reactor vessel.

17.3 Procedure Steps

*17.3.1 Attach the head lifting device upper sling assembly to the Polar Crane. Inspect the head lifting device, and remove any loose objects or dirt. Mark stud holes 12, 28, and 44 with yellow tape if not done as required earlier.

STOP

Do not start unless you intend not to leave Containment until the end of step 17.3.4. The following work cannot be left in mid-stream.

*17.3.2 Attach the sling assembly to the Reactor Vessel head lifting eyes. Lift the reactor vessel closure head off the storage stand about six (6) inches. Check levelness per step 11.3.5 above. (Refer to Figures 25, 28 and 29.)

*17.3.3 Raise the reactor head off its stand and move it in position over the reactor vessel. Align bolt holes 12, 28, and 44 in vessel head with the corresponding alignment pins and slowly lower the vessel closure head. (Verify that descent is even.) At about 170" off the flange check drive shaft to sleeve alignment. (See Figure 25.)

*17.3.4 Continue to slowly lower the head to about one (1) foot above the vessel and clean the reactor mating surfaces. QC Inspector, Watch Supervisor, and Maintenance Supervisor shall make joint inspection for cleanliness prior to close up. Lower head until it is seated on the reactor vessel. (See Figure 25.)

17.3.5 Remove the head lifting rig from the reactor vessel head and store it in the head storage stand.

17.3.6 Remove the lifting rig from the Polar Crane.

William J. Cahill, Jr.
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

June 15, 1978

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
ATTN: Mr. Victor Stello,Jr., Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

By letters dated October 16, 1975 and June 9, 1976, the Regulatory Staff requested various Indian Point Unit No. 2 information regarding the generic reactor vessel support issue. Partial responses to those requests were forwarded to the Staff by letters dated November 14, 1975, December 8, 1975, July 8, 1976, December 9, 1976, June 17, 1977 and March 9, 1978. The Indian Point Unit No. 2 reassessment has now been completed and is provided as Attachment A to this letter.

This submittal completes our response to the Regulatory information requests.

Very truly yours,

William J. Cahill

William J. Cahill, Jr.
Vice President

attach.

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ATTACHMENT A

Consolidated Edison Co. of New York, Inc.

Indian Point Unit No. 2

Docket No. 50-247

June 15, 1978

APPLICABILITY OF UNIT 3 ANALYSIS TO UNIT 2

In June 1977, Westinghouse submitted to the Nuclear Regulatory Commission a report on the integrity of the Indian Point 3 Nuclear Power Plant for postulated pipe breaks. This report, entitled, "Analysis of Reactor Coolant System for Postulated Loss-of-Coolant Accident: Indian Point Unit 3 Nuclear Power Plant," was submitted as WCAP 9117 (Proprietary) and WCAP 9130 (Non-proprietary). This report postulated pipe breaks at the locations in the primary loop which induce the most severe asymmetric loads on the reactor vessel. The analyses performed, as documented in the report, include the effect of modifications to the plant and demonstrate the adequacy of the entire system. The modifications proposed consist primarily of the addition of pipe motion limiters in the primary shield wall.

We have recently completed a study with Westinghouse in consideration of the applicability of the above mentioned Unit 3 reports to Indian Point 2. The two plants are very similar since much of the design of Unit 3 was duplicated from the Unit 2 design. Relevant similarities and differences between the two plants will be outlined in the following tabulated items. Discussion of the differences and the impact on the analytical conclusions will also be provided.

(1) The reactor vessels for the two units are identical with the exception of negligible differences in shell thickness which do not affect the analysis.

(2) The steam generators for the two units are the same model.

(3) The reactor coolant pumps for the two units are the same model.

(4) The reactor coolant piping layouts for the two units are structurally indistinguishable, that is the number of loops, the length of the pipes, the size of the pipes, the distance between components, etc.

(5) The internals (core support structures) configurations for the two units are identical.

(6) The CRDM's and CRDM supports for the two units are identical.

(7) The Unit 3 primary shield wall design was duplicated from the Unit 2 design.

(8) The vessel supports are identical.

(9) The steam generator supports are conceptually the same but have minor differences. Specifically, in Unit 3, the short pipe columns which extend from the steam generator foot to the main support frame were filled with grout, which increases the overall compression strength. This will have no effect on the bending of these columns which was the mode of interest in the Unit 3 analysis.

(10) The reactor coolant pump supports for the two units are conceptually the same. Parts of the vertical columns on the Unit 3 supports were reinforced. The very top sections of the columns (between the upper horizontal members and the pump feet) were the most significant parts of the pump supports in the Unit 3 analysis; only one of the three columns was reinforced in this area on Unit 3. In addition, the tie rod material for one tie

rod was changed on Unit 3. This particular tie rod receives minimal loads for a postulated break at the vessel nozzles.

(11) The fuel for the two plants is 15 x 15. However, the Unit 2 fuel assemblies have 9 grids vs. 7 grids in Unit 3 and the Unit 2 grids are stronger.

(12) The thermal and hydraulic design parameters of the two plants are very similar. Specifically the system nominal pressures are the same, the flows are the same, the vessel inlet temperature for Unit 2 is 0.4°F greater than Unit 3 and the vessel outlet temperature for Unit 2 is 4.4°F less than Unit 3.

(13) The two controlling case auxiliary lines for Unit 3, the residual heat removal (RHR) line attached to the hot leg of loop 32 and the accumulator line attached to the cold leg of loop 33, were analyzed. The geometric layout of these lines is identical for Unit 2 and Unit 3 but some differences do exist in their supports. The results of the Unit 2 piping support reliability improvement program have been considered in this evaluation and consequently, the results of the Unit 3 RHR line analyses are directly applicable to Unit 2. With regard to the accumulator lines, they have slightly different support mechanisms with the major difference being added vertical support in Unit 2 located outside the crane wall (approximately twenty feet from the primary pipe branch nozzle) near an elbow which produces a vertical drop in the accumulator line. Both units have a vertical hanger inside the crane wall (approximately ten feet from the branch nozzle) and a pipe restraint near the wall (approximately eighteen feet from the branch nozzle).

A summary follows which evaluates the similarities and differences of the two units with respect to: (a) the forcing functions, (b) the structural mathematical model, and (c) the expected results of the analysis.

(a) Forcing Functions

The MULTIFLEX and loop forcing functions will be essentially the same since the loop layouts, components, and internals are the same and the thermal and hydraulic design parameters (temperature and pressure) are very similar.

The cavity pressurization will be essentially the same since the primary shield wall (reactor cavity) design is the same and the design parameters are very similar.

Therefore, the forcing functions for Unit 3 are applicable to Unit 2.

(b) System Structural Model

The mathematical model of the system will be very similar. The model of the vessel and internals for the DARI-WOSTAS Code will be identical. The vessel support stiffness as determined by test for Unit 3 will be the same for Unit 2. The total loop stiffness (Figures 3-33b and 3-34 for the vessel inlet and outlet break in the Unit 3 report) will be considered further.

The elastic/plastic system model of the loop for Unit 3 (discussed in Section 3-8 of the Unit 3 report) will be the same for Unit 2 except for the minor differences in the pump and steam generator supports. The differences in the steam generator support will not affect the stiffness attributed to that structure and the differences in the pump support will slightly reduce the stiffness. The effect from both the steam generator and the pump supports will be only slightly smaller than in Unit 3 (the Unit 3 loop stiffness which comes from the pump and steam generator supports is shown in Figure 3-33a of WCAP 9117).

However, the total stiffness attributed to the loop in the Unit 3 analysis was the combination of the loop resistance and the effect of the proposed restraints in the primary shield wall. The restraint design planned for Unit 3 is also planned for Unit 2. Referring to Figure 3-33b of the Unit 3 report, note that the loop resistance stiffness is small with respect to the contribution to the total stiffness of the piping restraints which become active at approximately 0.3 inches. (The shield wall restraint stiffness will be the same for the two units.) In other words, variations in the loop resistance stiffness will not significantly affect the total overall resistance provided to the vessel by the combination of the loop and the shield wall restraints. The total resistance function applied to the vessel for Unit 2 will be slightly smaller up to approximately 0.3 inches of vessel motion and essentially the same above 0.3 inches.

(c) Results of Analysis

The results of the analysis will not differ significantly for Unit 2. The vessel motion will be essentially the same since the forcing functions will be similar and the total resistance stiffness coming from the vessel supports, loop effects, and the shield wall restraint will not be significantly different. The most important resistance to limiting the motion of the vessel is the restraints in the shield wall. With the response of the vessel being similar, the expected results for Unit 2 for the internals, the fuel and the CRDM's are not expected to be different from those for Unit 3. In fact, the fuel grid loads should be smaller since Unit 2 has a greater number of grids per assembly and the Unit 2 grids are stronger.

The vessel motion is imposed on the loop for the evaluation of the loop piping, components, and component supports. Since the imposed vessel motion will be essentially the same, the calculated stresses in the piping, the loads into the components at the nozzles, and the deformation induced in the supports of the components will be essentially the same. Although one of the three highly loaded pump support members is reinforced on Unit 3 but not on Unit 2, the Unit 2 pump supports have sufficient capacity to carry the required loads.

In the vessel supports, the Unit 2 response will be identical to that for Unit 3. In the pump and steam generator supports, the induced strain will be essentially the same since the response is deformation controlled. Considering the small strains, and hence the large margin, reported for Unit 3, the same conclusion of satisfactory support conditions would be determined for Unit 2.

Based upon: (1) the identical configuration of the RHR lines of Unit 2 and Unit 3, (2) the similarity of the Unit 2 accumulator line model to that used in the Unit 3 analysis with the only significant difference being in the location of the third piping restraint from the branch nozzles, (3) parametric variations on the location of the auxiliary piping supports of typical plants which have demonstrated that the maximum auxiliary piping stress generally occurs near the primary pipe (as verified by the Unit 3 analyses), and (4) the significant margin demonstrated in the Unit 3 analyses (Section 4-10 of WCAP 9117); the conclusions for the auxiliary lines presented in WCAP 9117 for Unit 3 also apply to Unit 2.

In this evaluation, a comparison of Unit 2 and Unit 3 has been made. The plants are very similar. The applicability of the analysis report prepared for Unit 3 to Unit 2 was discussed. Due to the similarity of the plants, the nature of the system response, and the fact that the modifications proposed for Unit 3 are also proposed for Unit 2, the conclusions stated for Unit 3 in the report are applicable to Unit 2. Specifically, those conclusions were that the plant will retain its structural configuration and have the capability of being safely shut down following a postulated pipe rupture occurring at the location in the loop which causes the greatest asymmetric loads in the system.

Westinghouse Electric Corporation has performed extensive analyses on this issue of asymmetric loads. The NRC has also performed independent analyses of the Unit 3 plant and has stated that the methods and results presented in the report by Westinghouse are conservative. The specific analysis on Unit 3, the NRC independent verification and statement of conservatism on that analysis, and the comparison of Unit

2 and Unit 3 presented in this evaluation, demonstrate that the conclusions drawn from the Unit 3 analysis are applicable to Unit 2 and satisfy the Regulatory requests with regard to Unit 2.

Docket
~~50-244~~
50 247

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 21, 1978

All Power Reactor Licensees

Gentlemen:

SUBJECT: REVISIONS TO INTRUSION DETECTION SYSTEMS AND ENTRY CONTROL HANDBOOKS AND NUCLEAR SAFEGUARDS TECHNOLOGY HANDBOOK

Enclosed is a copy of the Nuclear Safeguards Technology Handbook which was prepared under contract for the Department of Energy (DOE). The purpose of this handbook is to convey an understanding of the current SS safeguards technology development program and its prospective relevance and use to U.S. industrial and utility organizations, as well as to other U.S. government agencies and international organizations.

Also enclosed are updates to the "Entry-Control Systems Handbook" and the "Intrusion Detection Systems Handbook" that were sent to you earlier.

Sincerely,

James R. Miller

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosures:
As stated

cc w/o enclosures:
Service List

Docket 50-247

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 12, 1978

All Power Reactor Licensees and Applicants

Gentlemen:

This letter and enclosed Sandia Barrier Technology Handbook, dated November 1977, are being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with applications for a license to operate or construct a power reactor.

The Barrier Technology Handbook is designed to provide state-of-the-art information on the role of barriers in security and is provided as a reference document. Feedback on this handbook is encouraged from all recipients and should be addressed to Dr. Samuel C. T. McDowell, Assistant Director for Research and Development, Division of Safeguards and Security, DOE.

Sincerely,

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosure:
Barrier Technology Handbook

cc w/o enclosure:
Service List

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~~8009090822~~