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TECHNICAL EVALUATION REPORT

CONTROL OF HEAVY LOADS — PHASE II

CONSOLIDATED EDISON COMPANY

INDIAN POINT NUCLEAR POWER PLANT UNIT 2

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

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1. INTRODUCTION

1.1 PURPOSE

This technical evaluation report documents a review of load handling equipment operated in the vicinity of spent fuel and equipment employed for reactor shutdown and fuel element decay heat removal at Indian Point Nuclear Power Plant Unit 2. This review constitutes the second phase of a two-phase review instituted to resolve a generic issue pertaining to the safe handling of heavy loads at nuclear power plants.

1.2 GENERIC BACKGROUND

Generic Technical Activity Task A-36 was established by the Nuclear Regulatory Commission (NRC) staff to systematically examine staff licensing criteria and the adequacy of measures in effect at operating nuclear power plants to ensure the safe handling of heavy loads and to recommend necessary changes in these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [1] to all power reactor licensees, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612 [2]. The staff concluded from this evaluation that existing measures to control the handling of heavy loads at operating plants provide protection from certain potential problems but do not adequately cover the major causes of load handling accidents and should be upgraded.

To upgrade measures for the control of heavy loads, the staff developed a series of guidelines to implement a two-part objective. The first part of the objective, to be achieved through the implementation of a set of general guidelines expressed in NUREG-0612, Section 5.1.1, was to ensure that all load handling systems at nuclear power plants have been designed and are operated so that their probability of failure is appropriately small for the critical tasks in which they are employed. The results of the reviews associated with this part of the staff's overall objective were provided in a series of technical evaluation reports identified as Phase I reports. The second part of the staff's objective, and the subject of this report, was to be achieved through guidelines expressed in NUREG-0612, Sections 5.1.2 through 5.1.5. The purpose of these guidelines was to ensure that, in the case of specific load handling systems used in areas where their failure might result in significant consequences, either (1) features have been provided, in addition to those required for all load handling systems, to make the potential for a damaging load drop extremely small or (2) conservative evaluations of load handling accidents indicate that the potential consequences of a load drop are acceptably small.

1.3 PLANT-SPECIFIC BACKGROUND

On December 22, 1980, the NRC issued a letter [3] to Consolidated Edison Company of New York (Con Ed), the Licensee for Indian Point Unit 2, requesting the review of provisions for handling and control of heavy loads, the evaluation of these provisions with respect to the guidelines of NUREG-0612, and the provision of certain additional information to be used for an independent determination of conformance to these guidelines. The results of this independent evaluation with respect to general load handling equipment and procedures (Phase I) were provided on December 29, 1981 [4]. On December 3, 1981, Con Ed provided an initial Phase II report [5] concerning conformance with staff guidelines for specific load handling systems operated in areas where a load drop might result in significant consequences. That report provided the basis for this technical report.

2. EVALUATION

This section presents an evaluation of critical load handling areas at Indian Point Unit 2. Separate subsections are provided to identify the criteria used in this evaluation and each of the plant areas considered. For each such area, relevant load handling systems are identified, Licensee-provided information related to the evaluation criteria or proposed alternatives is summarized and evaluated, and a conclusion as to the extent of compliance, including recommended additional action or requirements for additional information as appropriate, is provided.

2.1 EVALUATION CRITERIA

The objective of this review was to determine if plant arrangements and load handling equipment design were such that either the likelihood of a load handling accident that could damage spent fuel or equipment used in reactor shutdown or fuel element decay heat removal is extremely small or that the consequences of such damage, should it occur, will be acceptable. Guidance contained in NUREG-0612, Sections 5.1.2, 5.1.3, and 5.1.5 (for pressurized water reactors) and in 5.1.4 and 5.1.5 (for boiling water reactors) forms the basis for the conclusions reached in this section and is briefly summarized as follows.

For a determination that the likelihood of damage is extremely small:

- The design of the load handling system (i.e., crane or hoist and underhook lifting devices) is consistent with, or equivalent to, the NRC staff criteria for single-failure-proof cranes identified in NUREG-0554 [6], or

- The plant physical arrangement is such that a crane operated in the vicinity of spent fuel or safety-related equipment is prevented from travelling to a position from which a load drop can be expected to damage such equipment.

For a determination that the potential consequences of damage following a load drop will be acceptable:

- In the case of potential damage to spent fuel, calculations have been provided to demonstrate that potential radiological doses at the site boundary will not exceed 25% of the limits specified in 10CFR100 and that the post-accident configuration of the fuel will not result in a K_{eff} larger than 0.95.

- In the case of damage to the reactor vessel or spent fuel pool, it can be demonstrated that this damage will be limited to the extent that the fuel will not become uncovered.

- In the case of damage to equipment or components employed for reactor shutdown or fuel element decay heat removal, it can be demonstrated that the safety-related function of the affected system will not be lost.

2.2 OVERHEAD HANDLING SYSTEMS

2.2.1 Summary of Licensee Statements and Conclusions

The Licensee identified the following load handling systems, capable of carrying heavy loads over the indicated areas, to be subject to the Phase II criteria of NUREG-0612:

1. In the vicinity of the spent fuel pool:

    - spent fuel storage building crane

2. In the vicinity of the reactor vessel:

    - containment polar crane

3. In the vicinity of equipment required for safe shutdown:

    - containment polar crane
    - auxiliary feedwater pump building monorail
    - PAB component cooling water pump monorail spur track
    - diesel generator building overhead hoist.

The weight of a heavy load is noted by the Licensee to be 2300 lb or greater.

2.2.2 Evaluation and Conclusion

The Licensee's evaluation of those handling systems subject to compliance with Phase II of NUREG-0612 is consistent with the conclusions of Reference 4.

2.3 SPENT FUEL POOL AREA

2.3.1 Spent Fuel Storage Building Crane

2.3.1.1 Summary of Licensee Statements and Conclusions

The Licensee stated that no heavy loads are currently handled by the 40/5-ton fuel storage building crane, although evolutions involving a spent fuel cask are anticipated during 1984. At present, mechanical stops are provided on the crane rails to prevent travel of the crane over the spent fuel pit. Removal of these stops is procedurally controlled and requires the approval of the Operations Engineer. The only routine removal of these stops occurs during the movement of new fuel assemblies or other non-heavy loads (such as neutron source or burnable poison rods). For these reasons, the Licensee excluded this crane from further consideration. No heavy loads are routinely carried within the vicinity of the spent fuel pool. The Licensee stated that, following selection of a cask, the need for modification of the fuel storage building crane will be assessed.

2.3.1.2 Evaluation

The Licensee's response has been compared with the evaluation criteria of NUREG-0612, Section 5.1.2(2). The mechanical interlocks that have been installed to prevent movements of the fuel storage crane over the spent fuel pit (SFP) satisfy, to a large degree, the NUREG criteria for installation of interlocks. No consideration appears to have been given, however, to establishment of the 15-ft "buffer zone" from the edge of the SFP, designed to prevent a dropped load from damaging the SFP walls or tipping or rolling into the SFP. Although the Licensee stated that no heavy loads are routinely carried in the fuel storage building, movements of such loads are not specifically prohibited by technical specifications or physical restraints; it is therefore recommended that the location of the crane mechanical interlocks be modified to provide adequate physical separation between the load and the edge of the SFP. Present procedures that allow these interlocks to be bypassed with the approval of the Operations Engineer are consistent with NUREG-0612 because no heavy loads are moved in the SFP area while the interlocks are bypassed. Procedures should be modified, however, to require that appropriate analysis be performed and approved prior to bypassing interlocks which allow movement of a heavy load into the SFP area in accordance with NUREG-0612, Section 5.1.2(2)(e); if not, the Licensee should adhere to the recommendations of Section 5.1.2(3) for movements within the SFP area (i.e., 25-ft separation, isolation of hot spent fuel).

Review of Indian Point Unit 2 Technical Specification 3.8.7 has identified the following additional restraints imposed on load movements in the SFP area:

- o NRC approval of the cask handling system is required prior to movement of the spent fuel cask over any region of the SFP if the SFP contains spent fuel.
- o Any load whose weight is greater than the weight of a storage rack and handling tool shall not be moved on or above the 95-ft elevation in the fuel storage building (the top of the SFP).
- o No heavy load (as defined by NUREG-0612) shall be moved over spent fuel in the SFP.

The requirement that NRC approval of the cask handling system be obtained prior to cask movement is an approach which is consistent with NUREG-0612. However, as no information on a proposed system has been provided, an evaluation of the cask handling system must be deferred until the Licensee provides more definitive information.

Additional information is requested of the Licensee regarding the 95-ft elevation height requirement for lifting loads in the fuel storage building. First, it is requested that the weight of a storage rack and handling tool be identified in order to quantify its relationship with a heavy load as defined by NUREG-0612. Clarification is also requested to explain the selection of the 95-ft elevation as the upper limit for movements and whether this height is intended to control movements inside or outside the SFP. If it is intended to control load movements within the SFP and if the referenced weight is greater than that of a heavy load, then additional information should be forwarded to demonstrate compliance with the criteria of Section 5.1.2(3).

In addition, although the present restriction prohibiting movements of heavy loads over spent fuel in the SFP satisfies the Phase I criteria of

Interim Protection Measure 1, such a restriction does not provide the defense in depth that is intended by adherence to the recommendations of Section 5.1.2(3). Therefore, the Licensee should reevaluate compliance with NUREG-0612 or modify the technical specification to provide the spatial separation between load and target as well as the segregation of hot spent fuel specified in NUREG-0612, Section 5.1.2(3).

2.3.1.3 Conclusion and Recommendations

Measures taken at Indian Point Unit 2 to satisfy NUREG-0612, Phase II concerns in the area of the spent fuel pit are not fully consistent with the NUREG recommendations. Although mechanical stops have been installed and technical specifications are in place which provide some measure of control in and around the SFP area, such measures do not fully satisfy NUREG criteria. Therefore, the following actions are requested of the Licensee:

- o To prevent movements into the SFP area, the Licensee should either (1) modify the location of the mechanical stops to provide an adequate buffer around the SFP or (2) revise the technical specifications to prohibit the movement of any heavy load into the area without the performance of an analysis consistent with NUREG-0612, Appendix A.

- o For movements of heavy loads which must be made within the SFP area, the Licensee should (1) modify administrative procedures which allow the mechanical stops to be bypassed to require either an analysis consistent with NUREG-0612, Appendix A, or compliance with the separation and fuel segregation criteria of Section 5.1.3(3) or (2) revise the technical specifications to satisfy the separation and fuel segregation criteria of Section 5.1.3(3) of NUREG-0612.

- o Provide additional information regarding Technical Specification 3.9.7, including the weight of a storage rack and handling tool, as well as justification of the 95-ft elevation lift height restriction.

2.4 REACTOR VESSEL AREA

2.4.1 Containment Polar Crane

2.4.1.1 Summary of Licensee Statements and Conclusions

The 175/35-ton containment polar crane was identified as the only crane subject to NUREG-0612, Phase II criteria in the vicinity of the reactor vessel. The Licensee stated that a probabilistic failure analysis was performed for this crane applicable to removal and installation of the reactor vessel head (169 tons) and the upper internals (69 tons), which are the bounding load drops of concern. Results of this analysis indicated that the probability of dropping either load after initial lift-off and leveling (at a height of 1.5 ft for 15 minutes) is extremely small, but is slightly greater than the probability of a drop from a greater height. Therefore, load drop consequences have been evaluated based upon a load drop that occurs from 1.5 ft during initial lift-off and leveling. No credit or reliance has been placed on mechanical or electrical safety features, and the crane is not a single-failure-proof system.

One other load is noted by the Licensee to be carried over the open core: the 5-ton in-service inspection tool is lifted by the 35-ton containment polar crane auxiliary hoist. Adequate load handling reliability is assured for this lift because the hoist was designed to and fully satisfies current industry standards and was built with a factor safety of 5:1. Therefore, the safety factor for the load of concern is far greater than 10:1, which satisfies the intent of Section 5.1.6 of NUREG-0612.

Three consequences of load drops by the polar crane main hoist into the open reactor vessel were considered: (1) loss of vessel integrity, (2) radiological release from fuel cladding damage, and (3) a criticality condition from fuel crushing. Consequences 1 and 2 were analyzed for a load drop during initial lift and leveling, based upon the previously identified failure analysis.

To determine the effects of a load drop into the open core, an analysis was performed using the methodology of WCAP-9198, "Reactor Vessel Head Drop." Based upon limitations imposed by a drop height of 1.5 ft, significant impact loads are not expected; therefore, fuel damage is not predicted from a drop of the reactor vessel head. Evaluation of the total anticipated impact load versus allowable structural stresses indicates that a loss of nozzle integrity is not predicted; therefore, the reactor coolant pressure boundary will remain intact.

The limiting load drop for consideration of fuel damage is a drop of the upper internals during initial lifting. Based upon analysis of a drop from 1.5 ft, the fuel cladding will experience a total strain of 0.22%, which is far less that the 1% strain required to achieve cladding failure. Therefore, a fission product release from the fuel is not predicted.

To demonstrate that a criticality condition will not occur as a result of fuel crushing, the Licensee stated that the geometry of the Indian Point reactor is the same as that analyzed in NUREG-0612 and that therefore the maximum expected increase in K_{eff} would be about 0.02. Indian Point Unit 2 technical specifications require 10% $\Delta K/K$ during head removal and while loading and unloading fuel from the reactor. Thus, Criterion II of NUREG-0612, Section 5.1 is satisfied.

In addition to providing an analysis of load drops into the open reactor vessel, the Licensee stated that loads lifted while the reactor vessel head is installed were not considered. Such loads include the control rod drive mechanism (CRDM) missile shields (23 tons), the CRDM missile shield support beams, and the reactor vessel head stud tensioners. No administrative controls are deemed necessary because none of these loads may be carried over an open vessel. A number of other loads are also present which may be moved over the open reactor vessel; however, there are procedural controls that prohibit movement over the refueling cavity when the vessel head is removed and irradiated fuel is present in the vessel. These procedures are strictly enforced by individuals in charge and will be reviewed with operators during training.

2.4.1.2 Evaluation

The Licensee's proposed measures have been compared with the recommendations of NUREG-0612, Section 5.1.3(3) for loads handled in the vicinity of the reactor vessel. The Licensee identified the 5-ton inservice inspection tool to be handled by the 35-ton polar crane auxiliary hoist and justified the compliance of this hoist with NUREG-0612, Phase II through use of safety factors in excess of 10:1. It is agreed that use of safety factors indicated by the Licensee (i.e., greater than 10:1) will provide an increase in load handling reliability consistent with that provided through the use of single-failure-proof or redundant handling systems. Such a comparable approach has been accepted in evaluation of specific elements of the handling chain, as documented in Section 6 of ANSI N14.6-1978, and has been used to evaluate special lifting devices in similar applications. It is noted, however, that to completely address the issue of loads handled by the auxiliary hoist, the Licensee should address the following additional concerns:

- o Although one load (inservice inspection tool) has been identified to be carried over the reactor vessel, insufficient information has been provided to verify that other (possibly heavier) loads are not carried by this hoist or to evaluate measures implemented by the Licensee to preclude the auxiliary hoist from handling loads in the vicinity of the reactor vessel. It is recognized that in certain unique circumstances (specifically where the administrative controls provide large separations between the control limits and the impact area of interest that are readily monitorable and strictly enforced), administrative controls can be found, on the basis of engineering judgment, to provide a high degree of certainty that loads will never be carried over the target. The Licensee has not demomstrated that these restrictions exist or that their exception is appropriate.

- o Although it has been clearly established that the auxiliary hoist has sufficient factors of safety greater than the load identified, similar assurances have not been provided for lifting devices (slings or special lifting devices) or lift attachment points located on the load itself. Verification is requested of the Licensee to ensure that these items also have factors of safety in excess of 10:1.

Analyses which attempt to demonstrate compliance of the polar crane main hoist with Criteria I and III of NUREG-0612, Section 5.1 have been performed by the Licensee. Exception is taken, however, with the Licensee's assumption of a drop height of 1.5 ft. Such an assumption is not consistent with Appendix A of NUREG-0612, which specifies that load drop analyses should consider drops from the maximum height while on the guide studs and also at the maximum height achievable in order to estimate the consequences of such a drop. Although the Licensee stated that the probabilities of load drops during initial lift and leveling and while at maximum height are slightly different, it is reasonable to assume that the consequences of a drop from the maximum height (greater than 29 ft) will be significantly different than the consequences of the 1.5-ft load drop analyzed by the Licensee. The analytical approach taken by the Licensee appears to reflect a misunderstanding of the fundamental issue of NUREG-0612. NUREG-0612, Appendix B, acknowledges that the probability of a load drop from a high quality commercial crane is relatively low. This recognition forms the basis for allowing fairly high consequence criteria (1/4 of 10CFR100 limit on radiological dose, for example). It was the intent of NUREG-0612 to provide an additional reduction in load handling system failure probability, approximately an order of magnitude, through the use of cranes consistent with NUREG-0554 where the specified consequence criteria could not be demonstrated to be satisfied through consequence analysis. Therefore, the approach taken by the Licensee is not consistent with NUREG-0612 and should be reevaluated. The method of analysis used by the Licensee should also be reevaluated as WCAP-9198 is not an approved NRC topical report in accordance with Reference 7.

For Criterion II, information provided by the Licensee indicates that the increase in reactivity due to crushing of fuel should be similar to that presented in Section 2 of NUREG-0612 based upon the Licensee's statement that assumptions and plant parameters at Indian Point Unit 2 are the same as the NUREG example. Therefore, in accordance with NUREG-0612, Appendix A, a value of 0.05 may be conservatively assumed as the maximum reactivity insertion value. Although the Licensee stated that technical specifications require at least 10% $\Delta K/K_{eff}$ during head removal and while loading or unloading fuel, insufficient information is available to ensure that such a condition exists at all times when heavy loads are handled while the vessel head is removed. Such assurances are necessary to ensure that a criticality condition caused by a load drop will not occur at times other than those specified by the technical specification when the head is removed. This additional information is required in order to establish that movements of heavy loads in the vicinity of the reactor vessel are consistent with Criterion II.

The Licensee's evaluation of other loads in the containment, with the head installed or removed, also do not appear to be consistent with NUREG-0612. It is not agreed that no administrative controls are required for loads carried with the vessel head installed. The Licensee appears to rely on the use of administrative controls to eliminate from further consideration certain heavy loads handled when the reactor vessel head is installed. In general, such procedural controls are not equivalent, in accordance with NUREG-0612 guidelines, to physical restraint or enhanced load handling system reliability in reducing the likelihood of a load drop over the reactor vessel. It is recognized, however, that in certain unique circumstances (specifically where the administrative controls provide large separations between the control limits and the impact area of interest that are readily monitorable and strictly enforced), administrative controls can be found, on the basis of engineering judgment, to provide a high degree of certainty that loads will never be carried over the target. The Licensee has not demonstrated that these restrictions exist or that their exception is appropriate.

2.4.1.3 Conclusions and Recommendations

Analyses and controls implemented in the reactor vessel area at Indian Point Unit 2 are not consistent with the Phase II guidelines of NUREG-0612. To conform to the NUREG criteria, the following Licensee actions are requested:

For the polar crane auxiliary hoist:

1. Justify the use of administrative controls to prevent the auxiliary hoist from carrying other loads in the vicinity of the reactor vessel.

2. Verify that lifting devices and lift attachment points have factors of safety consistent with that identified for the auxiliary hoist (i.e., 10:1).

For the polar crane main hoist:

1. Reperform load drop analysis in accordance with Appendix A of NUREG-0612 to include consideration of a load drop from the maximum height (Criteria I and II).

2. Verify that a margin of 10% $\Delta K/K_{eff}$ exists whenever any heavy load is moved during the entire period that the reactor vessel head is removed (Criterion II).

3. Justify the use of administrative controls to preclude the need for further analyses when the head is installed or removed.

2.5 OVERHEAD HANDLING SYSTEMS IN AREAS CONTAINING SAFE SHUTDOWN EQUIPMENT

2.5.1 Containment Polar Crane (Main Hoist)

2.5.1.1 Summary of Licensee Statements and Conclusions

Systems evaluations were performed by the Licensee to evaluate the consequences of load drops from the containment polar crane onto equipment required for safe shutdown in the containment. For purposes of analysis, the containment was subdivided into 10 separate regions, which were individually evaluated. The analyses evaluated scenarios based upon the reactor vessel head (RVH), both installed and removed, in addition to the following major assumptions:

- o All equipment in the region was conservatively assumed to be lost.
- o If reactor coolant system (RCS) piping was present in the region, a pipe break was assumed to occur.
- o Loss of component instrumentation in effect resulted in loss of the actual component (e.g., loss of steam generator level will result in loss of the steam generator).

The following cases were individually considered for each of the containment regions identified, where applicable. Fault tree analyses were performed to determine the availability of primary or backup cooling modes for each of the cases:

Case 1 - RVH removed (no RCS break)
2 - RVH installed (no RCS break)
3A - RVH removed (RCS break)
3B - RVH installed (small RCS break)
3C - RVH installed (large RCS break).

Analyses of the individual containment regions provided the following results. In Regions 3 (over the RHR heat exchangers) and 4 (RVH storage area), no possibility exists for unisolable RCS leaks; therefore, evaluations were conducted of Cases 1 and 2 only. In both instances, the Licensee was able to demonstrate the ability to maintain successful core cooling through the use of backup cooling modes.

The Licensee stated that no RHR or CCW equipment is present in Region 5 (over the reactor cavity) and therefore the primary cooling mode would not be lost. An independent ECCS analysis further indicated that, for damage to RCS piping in this region, the maximum level of flooding in the cavity and in the containment would not jeopardize any RHR or CCW components and the primary mode of core cooling in the cold condition would not be lost.

About Region 6 (four reactor coolant pumps [RCP[areas), the Licensee stated that adequate core cooling can be maintained following all postulated load drops. For Region 7 (operating deck area) it can be demonstrated in all but two cases that core cooling can be maintained. In the two cases noted, core cooling cannot be demonstrated by fault tree analyses due to potential drops disabling PORV piping adjacent to the pressurizer and the RHR injection line. However, due to the physical separation of the two components, the Licensee noted that it is extremely unlikely that a single load drop would disable both systems. Therefore, it is reasonable to assume that adequate means of core cooling will be present for all load drops in this region.

In Region 8 (steam generators), a breach of the steam generator shell would have no effect upon the ability to maintain core cooling in a cold plant condition. About Region 10 (slabs between the steam generators), the Licensee stated that potential consequences are bounded by the analysis for Regions 6 and 7; therefore, core cooling is maintained in all load drop cases. For Region 9 (instrument racks), analysis indicated that the ability to maintain core cooling is maintained following the loss of respective systems.

2.5.1.2 Evaluation

Analyses performed by the Licensee in the containment demonstrate that suitable redundancy or backup modes of decay heat removal exist to maintain adequate core cooling following any postulated load drop. Assumptions used by the Licensee in the performance of the analyses are conservative in nature and are consistent with those of Appendix A of NUREG-0612. Individual fault trees presented by the Licensee have been verified as accurate for the information presented.

Comparison of containment target areas with those regions established by the Licensee indicates that only one area, the immediate vicinity of the pressurizer, has not been addressed in the analyses of possible load drops. In addition, although the Licensee stated that a breach of the steam generator secondary side would not adversely affect cold shutdown capabilities, verification should be provided that RCS piping into the steam generators would not shear or that the consequences of such a break are acceptable.

2.5.1.3 Conclusion

Analyses performed by the Licensee demonstrate, with limited exceptions, that loads are handled by the polar crane main hoist in the vicinity of safe shutdown equipment in the containment with a degree of reliability consistent with NUREG-0612, Section 5.1.5. For the following exceptions, however, additional information is required to verify a comparable degree of reliability:

- o Verify that loads are handled in the immediate vicinity of the pressurizer in a manner consistent with Section 5.1.5 of NUREG-0612.
- o Verify that a load drop in the steam generator area will not breach RCS piping or demonstrate by appropriate analyses that the consequences are acceptable.

2.5.2 Containment Polar Crane (Auxiliary Hoist)

2.5.2.1 Summary of Licensee Statements and Conclusions

The Licensee stated that the 35-ton auxiliary hoist was evaluated for compliance with the single-failure-proof criteria of NUREG-0612, Section 5.1.6. The original design of the auxiliary hoist fully satisfies the criteria of CMAA-70 (1975) and ANSI B30.2-1976. Components are designed with a 5:1 design safety factor (ultimate strength), which implies that for loads of less than 17.5 tons, the design safety factor is 10:1, which satisfies the intent of Section 5.1.6 for increased safety factors in lieu of the normal 5:1 safety factors. Only one load which exceeds this 17.5-ton limit, the 25-ton equipment hatch door/airlock, was identified to be carried.

The Licensee performed an evaluation of the consequences of a load drop for the equipment hatch door. Fault tree analysis indicates that for a load drop in the area where the hatch door is carried, the ability to shut down the plant safely would not be lost with the reactor vessel head either in place or removed. Sufficient backup cooling systems are available to provide cooling to the reactor.

In addition to design to industrial standards and the satisfactory performance of a fault tree analysis, considerations are available which further reduce the likelihood of a load drop. Wire rope reeving of the auxiliary hoist consists of 7/8-in wire rope with a rated breaking strength of 245 tons, which approximates a 10:1 safety factor for a lift of the 25-ton equipment hatch door. Redundant 150% holding brakes are installed which are engaged when hoist power is lost or removed. The following actions will also be taken to satisfy the intent of NUREG-0612, Section 5.1.6:

1. Slings certified to ANSI B30.9 will be used with the auxiliary hoist for loads lifted in the annulus region.
2. An extensive inspection program, including thorough visual inspections prior to each refueling outage and f ctional checks, will be provided for brakes, limit switches, and ropes.
3. More stringent wire rope replacement criteria will be observed.
4. A second upper limit switch will be installed on the auxiliary hoist.
5. Load handling and operator qualification procedures have been upgraded to meet the guidelines of NUREG-0612 and ANSI B30.2-1976.

b. Evaluation

Although not in strict compliance with the criteria of NUREG-0612, Section 5.1.6 and Appendix C, for single-failure-proof handling systems, use of cranes that are designed in accordance with approved industrial standards to handle loads less than 50% of design capacity will provide safety margins in excess of 10:1. As noted in other standards used in the NUREG-0612 evaluation process (i.e., Section 6 of ANSI N14.6-1978), use of increased safety margins has been determined to be an acceptable approach and an alternative to modifications to existing cranes or provisions requiring the use of redundant hoisting trains. Therefore, for loads less than 50% of design capacity (17.5 tons for polar crane auxiliary hoist), such an approach satisfactorily meets the intent of NUREG-0612, Section 5.1.6 for the handling system.

However, it is noted that the issue of increased reliability of the remaining components of the lifting train (slings and attachment points) has not been addressed by the Licensee. Information should be provided which demonstrates that design safety margins similar to those provided by the crane (i.e., safety factors of 10:1 or use of redundant lifting devices) are also provided on related lifting devices and load attachment points.

For the single load noted to be greater than 50% of hoist capacity (25-ton equipment hatch door), the Licensee has performed systems analyses which provide reasonable assurances that the consequences of this load drop wil not preclude the ability to maintain core cooling.

2.5.2.3 Conclusion

Use of the polar crane auxiliary hoist in areas containing safe shutdown equipment partially satisfies NUREG-0612, Phase II criteria based upon (1) demonstration of factors of safety of 10:1 for loads less than 17.5 tons and (2) demonstration through systems analyses that a drop of the equipment hatch door will not preclude the ability to maintain core cooling. To fully satisfy Phase II measures for this hoist, however:

- o the Licensee should verify that similar design margins (safety factors of 10:1 or use of redundant lifting devices) are provided by lifting devices and load attachment points which connect the load to the hoist.

2.5.3 Auxiliary Feedwater Pump Building Monorail

2.5.3.1 Summary of Licensee Statements and Conclusions

The Licensee stated that the 5-ton auxiliary feedwater pump building (AFPB) monorail was also evaluated for compliance with the criteria of NUREG-0612, Section 5.1.6. Design of this monorail is in accordance with AISC specifications, which require a 5:1 design safety factor on ultimate strength. This represents a safety factor of 13:1 for the maximum loads that are anticipated to be moved by this monorail. No hoist is permanently attached; hoists with ratings at least twice the weight of the load and which meet ANSI B30.16 or equivalent industry standards will be used in order to achieve a safety factor of 10:1 for loads handled over the auxiliary feedwater pumps. Slings certified to ANSI B30.9 will also be used.

2.5.3.2 Evaluation and Conclusion

As noted in Section 2.5.2 fc. .he polar crane auxiliary hoist, use of safety margins of 10:1 is an acceptable alternative to verbatim compliance with NUREG-0612, Section 5.1.6. However, as also noted in Section 2.5.2, additional information is required to demonstrate that lifting devices and load attachment points connecting the load to the hoist are selected or designed based upon restrictions similar to those imposed on the hoist.

2.5.4 PAB Component Cooling Water Pump (CCWP) Monorail Spur Tank

2.5.4.1 Summary of Licensee Statements and Conclusions

The Licensee stated that the 7-ton CCWP monorail has been evaluated for compliance with the criteria of NUREG-0612, Section 5.1.6. Design of the monorail and attaching hardware is in accordance with AISC specifications, which require a design safety factor of 5:1. This represents a 45:1 safety factor for the maximum load to be lifted by this monorail. The hoist, designed to manufacturer's specifications, requires the same design factor of 5:1 and also has a 45:1 safety factor for the maximum load lifted. Slings certified to ANSI B30.9 will also be used.

2.5.4.2 Evaluation and Conclusion

As noted in Section 2.5.2 for the polar crane auxiliary hoist, use of safety margins of 10:1 is an acceptable alternative to verbatim compliance with NUREG-0612, Section 5.1.6. However, as also noted in Section 2.5.2, additional information is required to demonstrate that lifting devices and load attachment points connecting the load to the hoist are selected or designed based upon restrictions similar to those imposed on the hoist.

2.5.5 Diesel Generator Overhead Hoist

2.5.5.1 Summary of Licensee Statements and Conclusions

The Licensee stated that load drops from this hoist will not affect the ability to accomplish and maintain safe shutdown. Suitable redundancy of electrical power sources is present using either offsite power or one of the three emergency power gas turbines. All sources are independent of the diesel generator building and are unaffected by loss of the diesel generator units.

2.5.5.2 Evaluation and Conclusion

It is agreed that a load drop onto the diesel generators will not affect the plant's ability to maintain a safe shutdown. Therefore, the diesel generator overhead hoist may be exempted from further consideration by NUREG-0612.

3. CONCLUSION

This summary is provided to consolidate the results of crane-specific evaluations presented in Section 2. It is not meant as a substitute for the specific conclusions reached in the various subsections of Section 2, but rather is provided to allow the reader to focus on the key topics which should be addressed when resolving issues where the degree of load handling reliability provided by cranes at Indian Point Unit 2 was not found to meet the Phase II objectives of NUREG-0612. This section addresses those issues for which the information provided by the Licensee was insufficient to support a definitive conclusion and those issues for which the information provided by the Licensee has been evaluated to be an approach inconsistent with the guidance of NUREG-0612.

3.1 INFORMATION ISSUES

The information provided by the Licensee is either incomplete or insufficient to support an independent conclusion that load handling reliability is consistent with the evaluation criteria of Section 2.1 in the following areas:

- Loads Handled Outside the Spent Fuel Pool Area (Section 2.3.1.3)

  The Licensee should modify the location of the mechanical stops, if necessary, to provide an adequate "buffer zone" around the spent fuel pool to prevent a heavy load from tipping or falling into the pool if dropped. Verification should also be provided to ensure that appropriate analyses will be performed for any heavy load which must be handled in the spent fuel pool area.

- Loads Handled Inside the Spent Fuel Pool Area (Section 2.3.1.3)

  The Licensee should revise administrative procedures to require compliance with Section 5.1.2 of NUREG-0612 whenever mechanical stops are bypassed (i.e., appropriate load drop analysis or separation and segregation of spent fuel assemblies).

- Spent Fuel Pool Area Technical Specifications (Section 2.3.1.3)

  The Licensee should provide additional information to justify restrictions of Technical Specification 3.9.7, including identification of the weight of a storage rack and handling tool, as well as the limitations of the 95-ft elevation lift height restriction.

- Reactor Vessel Area - Criticality Considerations (Criterion II) (Section 2.4.1.3)

  Although the Licensee has documented that suitable margins to criticality exist during the vessel head removal replacement and fuel handling evolutions, similar assurances should be provided for handling of heavy loads at times other than those identified with the vessel head removed.

- Single-Failure-Proof Handling Systems (Sections 2.4.1.3, 2.5.2.3, 2.5.3.3, and 2.5.4.3)

  Use of several handling systems has been justified (polar crane auxiliary hoist, AFW building monorail, and PAB component cooling water monorail) on the basis that safety margins of greater than 10:1 exist between the margins designed into the hoist and the maximum weight handled. However, insufficient information has been provided by the Licensee to verify that other links in the load handling chain (i.e., lifting devices and lift attachment points) have similar margins. Such assurances should be provided to conclude that these systems meet the intent of Section 5.1.6 of NUREG-0612.

- Loads Handled by the Polar Crane near Safe Shutdown Equipment (Section 2.5.1.3)

  Verify that loads handled near the pressurizer satisfy Section 5.1.5 of NUREG-0612, and that a load drop in the steam generator area will not reach the RCS piping (or demonstrate that consequences of such a drop are acceptable).

3.2 APPROACH ISSUES

This review has revealed the following issues wherein the approach or position taken by the Licensee, based on information provided thus far, is inconsistent with the staff's objectives as expressed in the evaluation criteria of Section 2.1.

- Reactor Vessel Area Load Drop Analyses--Criteria I and II (Section 2.4.1.3)

  Load drop analyses performed by the Licensee are not consistent with those specified in Appendix A of NUREG-0612. Worst-case consequences of a major load drop onto fuel or the reactor vessel do not appear to have been determined based upon a maximum drop height. Therefore, analysis should be performed to include consideration of a drop from the maximum height.

- Use of Administrative Controls in the Vicinity of the Reactor Vessel (Section 2.4.1.3)

  The Licensee appears to rely on the use of administrative controls to eliminate from further consideration certain heavy loads handled in the vicinity of the reactor vessel. In general, such procedural controls are not equivalent, in accordance with NUREG-0612 guidelines, to physical restraint or enhanced load handling system reliability in reducing the likelihood of a load drop over spent fuel. It is recognized, however, that in certain circumstances (specifically where the administrative controls provide large separations between the control limits and the impact area of interest that are readily monitorable and strictly enforced), administrative controls can be found, on the basis of engineering judgment, to provide a high degree of certainty that loads will never be carried over the target. The Licensee has not demonstrated that these restrictions exist or that their exception is appropriate.

3. REFERENCES

1. V. Stello (NRC)
   Letter to All Licensees
   Subject: Request for Additional Information on Control of Heavy Loads Near Spent Fuel
   May 17, 1978

2. NRC
   NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"
   July 1980

3. D. G. Eisenhut (NRC)
   Letter to All Operating Reactors
   Subject: Control of Heavy Loads
   December 22, 1980

4. FRC
   Technical Evaluation Report, "Control of Heavy Loads at Indian Point Station Unit 2"
   TER-C5506-96, December 29, 1981

5. John D. O'Toole (Con Ed)
   Letter to D. G. Eisenhut (NRC)
   Subject: Control of Heavy Loads (Phase II)
   December 3, 1981

6. NRC
   NUREG-0544, "Single-Failure-Proof Cranes at Nuclear Power Plants"
   May 1979

7. NRC
   Topical Report Review Status
   July 15, 1983

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September 7, 1984

Docket No. 50-334

Mr. J. J. Carey, Vice President
Duquesne Light Company
Nuclear Division
Post Office Box 4
Shippingport, PA 15077

Dear Mr. Carey:

SUBJECT: BEAVER VALLEY UNIT 1 - COMPLETION OF REVIEW, POST-ACCIDENT SAMPLING SYSTEM (NUREG-0737, II.B.3)

By letter dated August 31, 1982 you provided information on the post-accident sampling system (PASS) at Beaver Valley Unit 1. By letters dated April 15, 1983, June 5 and July 16, 1984, you provided additional information as results of our requests for additional information.

Our review of the Beaver Valley Unit 1 PASS is complete. We find that all of the eleven (11) criteria as presented in NUREG-0737 are met. Details may be found in the enclosed Safety Evaluation.

This completes our work on the subject issue.

Sincerely,

/s/JDNeighbors, for

Steven A. Varga, Branch Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

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