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New York Power Authority

J. Phillip Bayne
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June 2, 1983
JPN-83-49

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Control of Heavy Loads

References:
1. NRC letter, D.G. Eisenhut to All Operating Reactors, dated December 22, 1980.
2. "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, dated July 1980.
3. PASNY letter, J.P. Bayne to T.A. Ippolito, dated October 15, 1981 (JPN-81-82).
4. PASNY letter, J.P. Bayne to D. B. Vassallo, dated February 26, 1982 (JPN-82-25).
5. "Control of Heavy Loads," draft Technical Evaluation Report, Franklin Research Center, dated March 25, 1982.
6. Federal Register, Vol. 48, No. 50, dated March 14, 1983, pp. 10772 - 10776.

Dear Sir:

Reference 1 requested that the Autnority review heavy load-handling operations at FitzPatrick and required a two-phase submittal of evaluations of their conformance to the guidelines of NUREG-0612 (Reference 2).

The Authority completed the first phase of the review and submitted an evaluation to the NRC in October 1981 (Reference 3).

In February 1982 the Authority submitted its evaluation for the second phase of the review (Reference 4). This evaluation indicated that the postulated consequences of certain load drops would not, or might not, meet the guidelines of NUREG-0612. Hence, the Authority prohibited

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handling of these loads until a further evaluation could be conducted demonstrating that the likelihood of such drops is acceptably small or, alternatively, that the postulated consequences of such drops are acceptable. This evaluation has been completed. A summary of its results is included as an attachment to this letter.

In March 1982, Franklin Research Center, under contract to the NRC, completed a draft Technical Evaluation Report of the Authority's phase 1 submittal (Reference 5). This report identified a number of items which required further analysis or protective measures. The Authority discussed these items with the NRC and Franklin Research Center in a telephone conference on October 7, 1982. In that conference call, the NRC requested that a response be provided that would document certain agreements reached during the call. That response will be submitted by June 30, 1983.

As noted in the attachment, the evaluation of postulated drops of the reactor vessel head, steam separator assembly, shipping casks, or recirculation pump motor indicate that the probability of such drops, following the initial lift and hold of these loads, is below or comparable to the NRC's current core melt "safety goal" probability of 1.0×10^{-4} per reactor year (Reference 6). With the exception of shipping casks, the Authority considers the calculated probabilities of drops of these loads to be sufficiently low as to preclude the need for analysis or protective measures beyond those discussed in the attachment. Hence, the load handling restrictions imposed by the Authority in Reference 4 for the reactor vessel head, steam separator assembly and recirculation pump motor have been removed.

The Authority will prohibit the handling of shielded shipping casks over the spent fuel pool until measures are taken either to further reduce the probability of a cask drop or to acceptably minimize the consequences of a drop.

If you have any questions regarding this letter or the attachment, please contact Mr. J.A. Gray, Jr. of my staff.

Very truly yours,

J.P. Bayne

J.P. Bayne
Executive Vice President
Nuclear Generation

cc: Mr. J. Linville
Resident Inspector
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ATTACHMENT TO JPN-83-49

NEW YORK POWER AUTHORITY

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

EVALUATION OF HEAVY LOAD HANDLING OPERATIONS

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INTRODUCTION

In our February 26, 1982 submittal (Reference 1), the Authority stated that some lifts of heavy loads at the James A. FitzPatrick Nuclear Power Plant (JAF) required additional analyses to demonstrate compliance with NUREG-0612 criteria. Interim measures prohibiting lifts of the reactor vessel head, steam separator assembly, recirculation pump motor and shipping casks, as identified in our letter, were applied until these evaluations could be completed. This report supplements our prior response and documents the results of our consultant's evaluations.

The evaluations involved control of heavy loads during refueling and maintenance activities. The evaluations involving refueling activities addressed the potential drop of the reactor vessel head or the steam separator assembly. The other evaluations, involving maintenance activities, addressed potential drops of the recirculation pump motor or shipping casks as they would be carried by the Reactor Building Crane across the operating floor and down the southeast equipment hatch. All of these loads are handled by the Reactor Building Crane, which was evaluated against industry design standards for such cranes and lifting devices and found to be in compliance (Reference 2). Additionally, the procedural requirements of NUREG-0612 for operator training and qualification and for crane inspection, testing and maintenance have been met for handling of all loads.

As described below, the evaluations of the handling activities associated with the reactor vessel head, steam separator assembly and recirculation pump motor have demonstrated that the likelihood of dropping these loads is extremely small. Consequently, these load drop scenarios do not require additional analysis or protective measures. Therefore, the Authority has removed interim restrictions on the handling of the reactor vessel head, steam separator assembly and recirculation pump motor.

The Authority will prohibit the handling of shielded shipping casks over the spent fuel pool until measures are taken either to further reduce the probability of a cask drop or to acceptably minimize the consequences of a drop.

Reactor Vessel Head, Steam Separator and Recirculation Pump Motor Lifts

Lifts of the reactor vessel head, steam separator assembly and recirculation pump motor by the Reactor Building Crane have been analyzed on a probabilistic basis. The study identified and quantitatively analyzed, using fault tree methods, the potential mechanisms for drops of these loads. The study was performed in accordance with the following steps:

- Review of the Reactor Building Crane system and associated testing, maintenance, inspection, training and lift procedures for removal and installation of the reactor head, steam separator and recirculation pump motor.

- Event identification and fault tree construction--determination of all the ways that the Reactor Building Crane system could fail, including:

  (1) Structural failure while subjected to normal load conditions;

  (2) Structural failure due to excessive load;

    i) two-blocking event

    ii) load hangup event

  (3) Overspeed event--loss of hoisting or lowering capability coupled with loss of brakes.

- Qualitative analysis--find system failure modes and establish all single failure events leading to system failure.

- Probabilistic analysis:

  (1) Find sources of data and determine applicability to JAF load-handling operations;

  (2) Compute probability of the undesired load drop event;

  (3) Probabilistically rank basic events and system failure modes (i.e., conduct a sensitivity analysis);

- Develop conclusions, recommendations and results.

The undesired load drop event for the analysis was defined in terms of two individual events:

- Drop during removal
- Drop during installation

These two events generate identical load drop scenarios, with two exceptions:

- During installation, a two-blocking event would most likely occur above the reactor head laydown area. Hence, this scenario is not considered during installation.
- A reactor head or steam separator load hangup event could only occur during removal. Again, this scenario is not considered during installation.

During removal operations, the reactor head and steam separator are initially lifted only several inches. The lifting rigs are then visually inspected before further lifting. To account for these operations, the analysis was segregated into two types of potential load drops:

- o Drop during initial lift
- o Drop after initial lift

Table 1 summarizes the mean probabilities of a load drop during lifts of the reactor vessel head, steam separator assembly and recirculation pump motor. The results indicate that the dominant failure mechanisms for these lifts (excluding the recirculation pump motor, where the dominant failure mechanism is related to overspeed events) are those related to the occurrence of a load drop during the initial lift and hold. Because the initial lift height is limited to less than 18 inches, the postulated consequences of a load drop at this stage of a lift were found to comply with the evaluation criteria of NUREG-0612 (i.e., reactor vessel integrity is maintained and no fuel damage will occur). The mean probabilities of a load drop subsequent to the initial lift and hold are shown in Table 1. Considering the inherent conservatism of the model used to calculate these probabilities, these mean values are themselves considered to be conservative.

In addition, the ACRS (Reference 3) and the NRC staff (Reference 4) have recently discussed quantified safety goals in an effort to establish a preliminary total probability for a reactor core melt. Those discussions led to publication of a preliminary core melt safety goal probability of 1.0 X 10^{-4} per reactor year (Reference 5). It should be emphasized that the probabilities listed in Tables I and II are for a load drop only. The consequences of any load drop accident would be considerably less severe than those expected from a core melt accident.

In summary, the dominant failure mode for lifts of the reactor vessel head and steam separator assembly were found to occur during the initial lift and hold stage. The consequences of drops at this stage were found to comply with NUREG-0612 evaluation criteria. For lifts of the reactor vessel head, steam separator assembly and recirculation pump motor, the probability of failure subsequent to the initial lift and hold stage was determined to be sufficiently small as to preclude the need for further analysis.

Shipping Cask Lifts

For shipping cask lifts, the reliability analysis described above applies also. In this case, the probabilities of failure are shown in Table 2. Using both systems and structural analyses, the Authority 's consultant has evaluated lifts of the various shipping and spent fuel casks identified for possible frequent use. Based on the evaluations, it has been determined that, by restricting the size of casks to about 35 tons and the lifting height to about 6 inches, the postulated consequences of load drops onto the refueling deck at Elevation 369' comply with NUREG-0612 evaluation criteria. That is, while some local structural damage may occur (e.g. concrete scabbing), no gross structural failures are expected. The systems analysis also indicated that safe shutdown capability and core cooling would still be maintained.

In view of the probabilities of a shipping cask drop over the spent fuel pool, the Authority will prohibit movement of shipping casks over the pool until additional measures are taken to further reduce the probability of a cask drop or to acceptably minimize the consequences of such a drop.

TABLE 1

MEAN PROBABILITY OF LOAD DROP PER LIFT

Load Drop Scenario	Mean Probability	
	Drop During Initial Lift	Drop After Initial Lift
Reactor Vessel Head	1.8×10^{-4}	6.9×10^{-5}
Steam Separator Assembly	2.3×10^{-4}	6.8×10^{-5}
Recirculation Pump Motor	not relevant*	3.3×10^{-4}

* Probabilistically, the dominant failure mechanism leading to a drop of a recirculation pump motor is an overspeed event. Consequently, a drop of the motor during the initial lift and hold stage is not considered.

TABLE 2

MEAN PROBABILITY OF SHIPPING CASK DROP PER LIFT

(for casks weighing 35 tons or less)

Shipping Cask Drop Scenario	Mean Probability	
	Drop During Initial Lift	Drop After Initial Lift
Equipment Hatch	not relevant*	3.3×10^{-4}
Operating Floor	8.2×10^{-5}	6.9×10^{-5}

* Probabilistically, the dominant failure mechanism leading to a drop of a shielded shipping cask is an overspeed event. Consequently, a drop of a shipping cask during the initial lift and hold stage is not considered.

REFERENCES

1. PASNY letter, J. P. Bayne to Domenic B. Vassallo, February 26, 1982 (JPN-82-25), "Response to NUREG-0612, Report No. 2" (nine-month submittal).

2. PASNY letter, J. P. Bayne to Thomas A. Ippolito, October 15, 1981 (JPN-81-82), "Response to NUREG-0612, Report No. 1" (six-month submittal).

3. U.S. Nuclear Regulatory Commission, An Approach to Quantitative Safety Goals for Nuclear Power Plants, NUREG-0739, October 1980.

4. NRC Office of Policy Evaluation, Discussion Paper on Safety Goals, July 23, 1981.

5. Federal Register, Volume 48, No. 50, March 14, 1983, pp. 10772 - 10776.