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TECHNICAL EVALUATION REPORT

CONTROL OF HEAVY LOADS — PHASE II

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE UNITS 1 AND 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 Arkansas Nuclear One Units 1 and 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 Arkansas Power and Light Company (AP&L), the Licensee for Arkansas Nuclear One Units 1 and 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 21, 1981 [4]. On December 22, 1982, AP&L provided a final Phase II report [5] concerning conformance to 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 Arkansas Nuclear One Units 1 and 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:

- o 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
- o 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:

- o 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 IN SPENT FUEL POOL AREA

2.2.1 Identification of Overhead Handling Systems

a. Summary of Licensee Statements and Conclusions

The Licensee identified [5] the following load handling systems as being subject to Phase II criteria of NUREG-0612:

- fuel handling crane (L3)
- auxiliary fuel handling crane
- new fuel handling crane (2L35).

The fuel handling crane (Equipment No. L3) was manufactured by P&H Harnischfeger and is a 100-ton electric overhead traveling bridge crane with a 10-ton auxiliary hoist. A 2-ton auxiliary hoist is suspended, monorail-fashion, from a 12-inch I-beam welded between the end trucks at the south end of the fuel handling crane (L3) and is designated as the Unit 1 auxiliary fuel handling crane. The Unit 2 new fuel handling crane (Equipment No. 2L35) was manufactured by Heco-Pacific Manufacturing Company and is a 4-ton capacity, top-riding, single-girder crane consisting of a bridge, monorail, trolley, and hoist.

The weight of a heavy load is noted by the Licensee to be any load weighing more than 2000 lb.

None of the identified cranes are in complete compliance with NUREG-0612, Section 5.1.6 (single-failure-proof) or in partial compliance supplemented by suitable alternative or additional design features.

b. Evaluation and Conclusion

The Licensee's identification of load handling systems capable of carrying loads over the spent fuel pool (regardless of capacity or load carried) is consistent with Phase II of NUREG-0612.

2.2.2 Evaluation of Phase II Compliance

a. Summary of Licensee Statements and Conclusions

The Licensee selected Alternative 4 from those identified in NUREG-0612, Section 5.1.2 for the evaluation of identified cranes. However, no analysis, as stipulated in Alternative 4, has been performed. The Licensee believes that the likelihood of a load drop from the fuel handling crane (L3) and the Unit 1 auxiliary fuel handling crane that would result in an unacceptable radioactivity release is extremely small for the following reasons:

1. Plant technical specifications prohibit the handling of loads in excess of 2000 lb over spent fuel, and plant procedures have been revised to prohibit the storage of irradiated fuel in the vicinity of the fuel pool gates and around the periphery of the fuel pools.

2. A safety factor of 5 to ultimate was used for the design of the 100/10-ton fuel handling crane.

3. The crane's box girders, as well as the crane girders and their support structures, were designed to resist both design basis and maximum credible earthquake forces. The crane box girders were also designed to resist vertical earthquake-induced loads with the lifted load of 100 tons. The crane girders and support structures were also designed to resist tornado wind loadings.

4. The crane is inspected in accordance with ANSI B30.2-1976 prior to its use, and the slings used in load handling operations with this crane comply with ANSI B30.9-1971 requirements.

5. The reinforced concrete slab on the bottom of the spent fuel pools in both units has been analyzed for a pool gate load drop in accordance with an impact load analysis methodology contained in ASCE's Report on Engineering Practice No. 58, "Structural Analysis and Design of Nuclear Plant Facilities," 1980. This analysis revealed that, although a fuel pool gate would slightly penetrate the bottom slab in the fuel pool, the slab would neither spall nor fail in shear. It is believed that this analysis is conservative and therefore an acceptable method of excluding a fuel pool gate drop from further consideration.

The Licensee also believes that the crane's hoist load block can be excluded from further consideration for the following reasons:

1. The main hoist is only used to lift a spent fuel cask or a fuel handling machine (if necessary) and other miscellaneous loads that are less than 2000 lb. Plant procedures identify load paths that do not cross over spent fuel in the pool.

2. Before use, the crane is inspected per ANSI B30.2, and all slings used with this crane meet ANSI B30.9 requirements.

3. The only crane failure that could cause the main hoist block to fail and thus fall on spent fuel in the pool would be a "two-blocking" event, which cannot occur with the hoist not in use.

4. The main hoist load block is considered an integral part of the crane.

The Unit 2 new fuel handling crane (Equipment No. 2L35) is used to handle new fuel, new control components, and other miscellaneous loads, such as steel hatch covers at elevation 404 ft in the auxiliary building, that crane L3 cannot reach. The only loads handled over fuel are either new fuel, new control components, or loads less than 2000 lb. The Licensee believes that this crane should be excluded from further consideration under NUREG-0612 for the following reasons:

1. The crane was designed and constructed per CMAA-70.

2. The crane is inspected and operated per ANSI B30.2-1976, and all slings that are used with this crane are in compliance with ANSI B30.9-1976.

3. The load of a new fuel assembly and its handling tool for Unit 2 is approximately 1450 lb, which is 18% of the crane's 4-ton rated capacity and less than the 2000-lb limit of loads carried over spent fuel in the pool. A new control component weighs approximately 96 lb; a hatch cover weighs approximately 2300 lb but is not handled over the fuel pool.

b. Evaluation and Conclusion

The Licensee has not provided an analysis per NUREG-0612, Appendix A requirements, as stipulated in selected Alternative 4 of Section 5.1.2, to show that a postulated load drop into the spent fuel pool would not cause uncontrolled release of radioactivity in excess of the dose limits of

Criterion I. Instead, the Licensee has relied on technical specification restrictions and administrative controls on the movement of loads over the spent fuel to conclude that the likelihood of load drop is extremely small. Reliance on safe load paths and other such adminstrative controls in lieu of mechanical stops or electrical interlocks does not necessarily ensure that the heavy loads will not be handled over the stored spent fuel due to operator inattention or error. The Licensee's discussion of crane features, other than seismic design, which make the likelihood of a load drop from the spent fuel area cranes extremely small addresses only features that conform to industrial standards and are consistent with the general requirements of NUREG-0612, Section 5.1.1. They are not extra features to improve the reliability of cranes' handling critical loads as discussed in Section 1.2 of this report. Technical specification restrictions on loads (not to exceed 2000 lb) to be carried over spent fuel stored in pools lend much more credibility to the Licensee's conclusions, and the Licensee has tried to take credit for these restrictions to demonstrate that a fuel handling accident would not result in potential offsite exposures in excess of dose limits specified in evaluation Criterion I of Section 5.1 of NUREG-0612. The Licensee, however, has not addressed the concern with regard to the criticality aspects of Criterion II following a postulated heavy load drop accident.

In conclusion, it appears that the Licensee has attempted to show conformance to the Guidelines of NUREG-0612, Section 5.1.2 through a combination of load drop analysis (for damage to the fuel pool subsequent to a gate drop) and a technical specification (for damage to spent fuel). Although the use of a technical specification may be found to provide protection equivalent to mechanical stops or electrical interlocks, as discussed in NUREG-0612, Section 5.1.2(2), such a determination cannot be made on the basis of the information provided. In order to be found equivalent to electrical interlocks in reducing the probability of a load drop, the associated technical specification should preclude carrying heavy loads a reasonable distance away from the stored fuel so that the load cannot be carried over spent fuel as a result of operator inattention or control system failure and so that the fuel will not be impacted following a handling system failure,

including an unfavorable handling device failure when the crane is operating in the area permitted by the technical specification. Further, the technical specification should be easily enforceable through markings or other provisions that allow an immediate determination of whether a technical specification limit has been exceeded. The Licensee should provide additional information concerning technical specification limits and enforcement provisions to allow evaluation of this alternative.

2.3 OVERHEAD HANDLING SYSTEMS IN REACTOR VESSEL AREA

2.3.1 Identification of Overhead Handling Systems

a. Summary of Licensee Statements and Conclusions

The Licensee identified the following load handling systems as being subject to Phase II criteria of NUREG-0612:

- o polar cranes (L2 and 2L2)
- o control rod drive (CRD) and general maintenance crane (L21).

The polar cranes at Arkansas Nuclear One Units 1 and 2 are P&H Harnischfeger 150/25-ton capacity, double-gantry circular cranes. The equipment numbers for these cranes are L2 and 2L2 for Units 1 and 2, respectively.

The Unit 1 CRD and general maintenance crane (Equipment No. L21) was manufactured by Heco-Pacific Manufacturing Company and is a 2-ton capacity, top-riding, single-girder crane consisting of a bridge, monorail, trolley, and hoist. The Licensee believes that the CRD and general maintenance crane (L21) in the Unit 1 reactor building can be excluded from further consideration for the following reasons:

1. Although the crane has a rated capacity of 2 tons, its maximum lifted load is a CRD mechanism whose total assembly weight is 935 lb. This crane also is used to move reactor vessel studs which weight 640 lb, reactor vessel head insulation pieces with an average weight of approximately 400 lb, portions of reactor vessel head cooling duct work whose maximum calculated weight is approximately 800 lb, and several other small miscellaneous loads. None of these loads is a heavy load since their respective weights do not exceed 2000 lb.

2. This crane is used to assist in several maintenance operations prior to the removal of the reactor vessel head. However, administrative controls are being developed to ensure that, prior to the removal of the reactor vessel head, the crane is locked in a position at the east end of the refueling canal so that it is incapable of carrying any load over the open reactor vessel. Administrative controls are being developed to ensure that it is also locked in this position and seismically restrained during normal plant operations.

   Also, administrative controls are being developed to ensure that the trolley and hoist are removed from the crane gantry and stored elsewhere during normal plant operations since there are no seismic restraints on the trolley.

b. Evaluation and Conclusion

The Licensee's identification of load handling systems capable of carrying loads (regardless of capacity or load) over the reactor vessel is consistent with NUREG-0612.

The Licensee's justifications for the exclusion of the CRD and general maintenance crane (L21) meet the intent of NUREG-0612 contingent upon implementation of intended administrative controls.

2.3.2 Evaluation of Phase II Compliance

a. Summary of Licensee Statements and Conclusions

No analysis was performed in accordance with the guidelines of NUREG-0612, Appendix A to demonstrate compliance with Criteria I through III. However, a head drop analysis from 3 ft 6 in above the vessel flange, performed by Babcock and Wilcox in November 1972, revealed that the support skirt assembly will be generally overstressed in compression and will yield moderately or buckle. A total displacement was estimated to be 7/16 in. This analysis is conservative since it was assumed that the head was a rigid falling weight and no attempt was made to evaluate its own elasticity which would have reduced the stresses even further.

The Unit 1 and Unit 2 polar cranes (Equipment Nos. L2 and 2L2, respectively) are rated at 150 tons for their main hooks and 25 tons for their auxiliary hooks. Both cranes were designed to support the maximum loads

caused by a 600-ton construction lift as well as normal operating loads plus earthquake forces resulting from a design basis earthquake.

In summary, it is believed that the likelihood of a load drop from the polar cranes that would result in an unacceptable radioactive release is extremely small for the following reasons:

1. A safety factor of 5 to ultimate was used for the design of the 150/25-ton polar cranes.

2. The box girders and the crane girder and its brackets were designed for a 600-ton (605 tons on Unit 2) load, and the maximum normal load is only 25% of this on Unit 1 (24% on Unit 2).

3. The polar cranes are inspected in accordance with ANSI B30.2-1976 only during outages. All polar crane inspections (except daily inspections) are performed under plant surveillance test programs when they are due. Surveillance that is due during plant operation will be performed at the beginning of each outage prior to the use of the crane. This includes periodic, quarterly, semiannual, and annual inspections. Plant procedures are being revised to ensure that the daily (frequent) polar crane inspections are performed prior to the use of these cranes. Exceptions to this are short-lived outages that do not require use of polar cranes. Slings used in lifting heavy loads will be in compliance with ANSI B.9-1971, and crane operators are trained and qualified in accordance with ANSI B30.2-1976.

4. The cranes were designed to resist earthquake forces generated by the design basis earthquake and the maximum credible earthquake.

The in-service inspection (ISI) tools are attached to the polar crane auxiliary hook and are considered as individual loads. The ISI tool is used for the reactor vessel inspection and the upper nozzle inspection on Unit 1 with fuel in the core and the upper plenum assembly removed. The ISI tool has design features that preclude the inadvertent contact of the tool's mast and remote arm with the irradiated fuel in the vessel core. This prevents an accidental criticality of the fuel which could be caused by the fuel being crushed. The ISI tool is not used on Unit 2 with fuel in the reactor. Based on the above considerations, the Licensee believes that the main hoist load block on polar cranes L2 and 2L2 should be excluded from further consideration as a heavy load.

b. Evaluation and Conclusion

The Licensee stated that the likelihood of a drop from polar cranes is extremely small because of the safety factors involved in the design of the cranes as well as improved inspection, maintenance, and operation procedures. The provisions are consistent with the general requirements of NUREG-0612, Section 5.1.1. They are not extra features to improve the reliability of the cranes handling critical loads as discussed in Section 1.2 of this report.

For the load drop scenario analyzed by Babcock and Wilcox, the structural integrity of the reactor vessel is apparently maintained, although local overstressing and buckling of support skirt occurs following a reactor vessel head drop from a height of 3.5 ft. The Licensee, however, did not provide the rationale for selection of the drop height employed in the reactor vessel impact analysis. The staff's intent, as expressed in NUREG-0612, Appendix A, was to ensure that an analysis relied upon to demonstrate the acceptability of the consequences of a load drop properly considered the maximum height from which the load might be dropped. The Licensee should provide justification for the selection of a 3-ft 6-in drop height including an indication and justification of the margin between this height and the maximum height that the reactor vessel head will be carried over the vessel. The margin selected should be reasonable and should accommodate both operator response and the ability of the operator or supervisor to ensure that the margin is not exceeded.

In addition, the Licensee's load drop analysis has not directly addressed whether or not Criteria I through III of NUREG-0612, Section 5.1 will be satisfied following the postulated load drop. Although it may be reasonable to assume that an accident that results in reactor vessel displacement of 7/16 inch will not result in fuel damage sufficient to cause an excessive radioactivity release or increase in reactivity, such conclusions should be drawn by the Licensee on the basis of its analysis and cannot be arrived at independently on the basis of the information provided.

2.4 OVERHEAD HANDLING SYSTEMS IN AREAS CONTAINING SAFE SHUTDOWN EQUIPMENT

2.4.1 Summary of Licensee Statements and Conclusions

The Licensee has provided detailed information in matrix form which identifies all handling systems, locations, and loads that may affect equipment or components required for safe shutdown; this information is provided in Appendix A. Discussions have been provided to eliminate load/impact area combinations (Hazard Elimination Category) based on separation and redundancy of safety-related equipment and other site-specific considerations. These considerations are summarized below.

Intake Structure Gantry Crane (L7)

The heaviest safety-related loads lifted by the intake structure gantry crane (L7) are the service water pumps, and it is not physically possible to drop one pump on top of another due to the separation of the pumps. This separation consists of thick, reinforced concrete walls between each pump bay in both Units 1 and 2. The Unit 1 motors are approximately 27 ft apart with 18-in-thick reinforced concrete walls separating them. The Unit 2 motors are approximately 8 ft apart with 18-in-thick reinforced concrete walls separating them. In addition, the size of the motors allows them only to be lifted through the roof hatch directly above them.

At present, there are no administrative controls to prevent the handling of other loads at the intake structure, such as the circulating water pumps, over safety-related equipment in this area. Intake structure crane procedures are being revised to incorporate these restrictions.

Fuel Handling Crane (L3)

Other than the spent fuel shipping cask, whose load drop has already been analyzed, the fuel handling machine is the heaviest load lifted by the fuel handling crane (L3). This load weighs less than the 100-ton shipping cask, and the slab at elevation 404 ft 0 in has been shown to survive a cask drop; therefore, the crane can be eliminated from further consideration because a load drop would not prevent safe reactor shutdown [7].

It should be noted that there are no safe shutdown components beneath the load paths of the remainder of the heavy loads handled by the fuel handling crane. The crane load block has been previously excluded from further consideration based on a discussion in Section 2.2.2.a.

Polar Cranes (L2 and 2L2)

The Unit 1 and Unit 2 polar cranes (Equipment Nos. L2 and 2L2) can be eliminated from further consideration due to separation and redundancy of safety-related equipment.

In Unit 1, the heaviest load that would be lifted during any plant condition other than cold shutdown would be the reactor vessel missile shields. The missile shields are normally lifted when the plant is in cold shutdown. Lifting these missile shields during hot shutdown would be extremely unusual. The worst-case load drop of a missile shield would affect a portion of decay heat removal system piping at the west end of the refueling canal at elevation 354 ft 0 in. This postulated case has been excluded from further consideration because the piping in question is routed against the outside face of the secondary shield wall and the load would have to slide down the shield wall in order to destroy this piping. Due to the physical orientation of the missile shield, it has been determined that this is an incredible scenario.

When the plant is in a cold shutdown condition or in a refueling shutdown condition, the remainder of the heavy loads listed in Appendix A may be lifted. There are only four loads whose load drop could affect the ability to maintain the plant in a cold shutdown condition, i.e., maintain decay heat removal capability. These are the reactor vessel head, the reactor coolant pump motor, pump, and structural support beam. The load paths of the reactor coolant pumps located in the north cavity pass over one decay heat removal line located near the reactor building sump. A load drop onto this line would not result in the loss of decay heat removal capability because one decay heat loop would still be available. The load paths of the reactor coolant pumps located in the south cavity pass over the "A" core flood line on the west side of the secondary shield wall, but a load drop on this line would not result in

the loss of decay heat removal capability. Finally, the load path of the reactor vessel head passes over the core flood line and the decay heat removal line for the "A" loop. Even if these lines were destroyed by a head drop, the redundant decay heat removal loop could still maintain core coverage.

In Unit 2, the heavy loads that might be lifted by the polar crane in a plant condition other than cold shutdown would be the vessel head stud stand or the refueling canal seal plate lifting rig. They are normally lifted when the plant is in cold shutdown. Although these loads pass over safety injection piping, the possibility of these loads penetrating several feet of reinforced concrete is extremely remote. For this reason, the carrying of these loads under a plant condition other than cold shutdown can be eliminated from further consideration.

When the plant is in a cold shutdown or a refueling shutdown condition, the remainder of the heavy loads listed in Appendix A may be lifted. There are several loads that may be lifted whose load drop could affect the ability to maintain the plant in a cold shutdown condition, i.e., maintain decay heat removal (shutdown cooling) capability. These are the reactor vessel head lift, the reactor coolant pumps and their structural steel support beams, the head maintenance structure, the jib crane (2L45), the refueling machine and other refueling system components, and the refueling cavity seal plate. Of these loads, the consequences of a reactor coolant pump motor drop or a reactor vessel head drop would envelope the other postulated load drops.

A reactor vessel head drop over the refueling cavity could result in the loss of shutdown cooling and safety injection piping from the reactor vessel through the "B" hot leg. However, this would only occur if the head penetrated a 4-ft-0-in-thick concrete slab. This would <u>not</u> cause loss of decay heat removal capability from the reactor vessel because this line could be isolated and the other cooling loop could be used to maintain shutdown cooling.

A drop of the "A" reactor coolant pump motor over the reactor building sump area could result in the loss of shutdown cooling and safety injection piping into the "C" cold leg as well as one containment spray line. It is

still possible to maintain adequate core decay heat removal after the loss of this piping by placing the shutdown cooling system in the recirculation mode using the HPSI pumps and by isolating the affected safety injection line.

Main Steam Isolation Valve (MSIV) Bridge Crane (2L10)

The load lifted by the MSIV bridge crane (2L10) and its respective load/target combinations can be eliminated from further consideration because a load drop would neither prevent safe reactor shutdown nor prohibit continued decay heat removal. The reactor and the plant must be in a cold shutdown condition before a MSIV is removed so a load drop would not prevent safe reactor shutdown. The heaviest component of a MSIV lift is a cylinder stiffener section that weighs approximately 10,000 lb. It is postulated that this load, if dropped, would penetrate the north penetration rooms which contain piping and electrical conduit servicing the service water system, the emergency feedwater system, the fire water system to containment, the main feedwater system, the MSIVs, penetration room ventilation, and a containment HVAC radiation monitor. Since the loss of these systems or portions of them when the plant is in a cold shutdown condition will not prohibit continued decay heat removal, this crane can be eliminated from further consideration.

2.4.2 Evaluation and Conclusion

Information provided by the Licensee is insufficient to allow an independent determination that all equipment associated with plant shutdown or decay heat removal in the vicinity of paths followed by heavy loads have been evaluated. The Licensee should provide additional information showing the location of load paths and equipment, including cables and motor control centers, associated with plant shutdown and cooldown. Additional information is also needed to justify various hazard elimination categories in Appendix A.

3. CONCLUSIONS

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. It is provided to allow the reader to focus on the key topics which should be addressed in seeking to resolve issues where the degree of load handling reliability provided by cranes at Arkansas Nuclear One Units 1 and 2 was not found to meet the objectives of NUREG-0612. This section addresses issues for which the information provided is felt to be inadequate to support a definitive conclusion and issues wherein the information provided has been evaluated as proposing an approach inconsistent with the guidance of NUREG-0612.

3.1 INFORMATION ISSUES

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

Reactor Vessel Area Load Handling Systems (Section 2.3.2)

- o The Licensee should provide justification for the selection of the 3-ft 6-in drop height, including justification of the margin between this height and the maximum height that the reactor vessel head will be carried over the reactor vessel. The analysis should be performed in accordance with NUREG-0612, Appendix A and based on maximum fall with due allowances for operator errors.
- o The Licensee should indicate that provisions that are in effect to prevent exceeding the procedural maximum lift height.
- o The analysis performed should indicate satisfaction of evaluation Criteria I through III of NUREG-0612, Section 5.1.

Safe Shutdown Equipment Area Load Handling Systems (Section 2.4.1)

- o The Licensee should provide additional information showing the location of load paths and equipment, including cables and MCCs, associated with plant shutdown and cooldown.

- o The Licensee should provide additional information to justify various hazard elimination categories in Appendix A of this report.

3.2 APPROACH ISSUES

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

Spent Fuel Pool Area Load Handling Devices (Section 2.2.2)

- o The Licensee appears to rely on the use of technical specifications and administrative controls to eliminate from further consideration certain heavy loads handled in the vicinity of the spent fuel pool. 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 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.

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APPENDIX A

LOAD/IMPACT AREA MATRIX, ARKANSAS NUCLEAR ONE UNITS 1 AND 2

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Appendix A. Load/Impact Area Matrix, Arkansas Nuclear One Units 1 and 2

Crane	Location	Impact Elevation	Load	Impact Area	Safety-Related Equipment	Hazard Elim. Category*
Fuel Handling Crane (L3)	Reactor Auxiliary Bldg. (Fuel Handling Area)	404'-0	Spent Fuel Cask (25-Ton)	Control Room Roof	Relay Panels	E
		404'-0	Hatch Over Crane Bay	Hatch Frame	None	C
		386'-6	Fuel Trans. Tube Gate Valve	Fuel Tilt Pit Floor	None	C
		362'-0	New Fuel Elevator	Fuel Tilt Pit Floor	None	C
		404'-0	Fuel Hand. Machine	Control Room Roof	Relay Panels	E
		362'-0	Upender	Fuel Tilt Pit Floor	None	C
		354'-0	New Fuel Ship. Cont.	R. R. Bay Floor	None	C
		354'-0	New Control Comp.	R. R. Bay Floor	None	C
		354'-0	New Fuel Elements	R. R. Bay Floor	None	C
		354'-0	New Control Equipment	R. R. Bay Floor	None	C
		362'-0	Fuel Transfer Carriage	Fuel Tilt Pit Floor	None	C
		362'-0	Fuel Pool Divider Gates	Spent Fuel Pool Floor	Fuel Pool	E
		404'-0	Crane Load Block	Fuel Hand. Floor; Fuel Pool Floor; R. R. Bay Floor	Spent Fuel in Fuel Pool; Relay Panels in Control Room	D

*Explanation of Hazard Elimination Categories:

A = Crane travel for this area/load combination prohibited by electrical interlocks or mechanical stops.

B = System redundancy and separation precludes loss of capability of system to perform its safety-related function following this load drop in this area.

C = Site-specific considerations eliminate the need to consider load/equipment combination.

D = Likelihood of handling system failure for this load is extremely small (i.e., Section 5.1.6, NUREG-0612 satisfied).

E = Analysis demonstrates that crane failure and load drop will not damage safety-related equipment.

Appendix A (Cont.)

Crane	Location	Impact Elevation	Load	Impact Area	Safety-Related Equipment	Hazard Elim. Category*
New Fuel Handling Crane (2L35)	Reactor Auxiliary Bldg. Unit 2	404'-0 or 362'-0	New Fuel Assembly	Control Room Roof; Spent Fuel in Pool	Relay Panels in Control Room; Spent Fuel in Pool	E
		404'-0 or 362'-0	New Cont. Component	Control Room Roof; Spent Fuel in Pool	Relay Panels in Control Room; Spent Fuel in Pool	E
Intake Structure Gantry Crane (L7)	Intake Structure Unit 1	366'-0 or 378'-0	Unit 1 Service Water Pump Motor	Service Water Pump Room Floor or Intake Structure Roof	Pump Room	B
		322'-6 or 378'-0	Unit 1 Service Water Pump	Intake Structure Base Mat	Pump Room	B
		366'-0 or 378'-0	Motor Driven Fire Pump and Motor	Pump Room Floor or Intake Structure Roof	Pump Room Floor	B
		366'-0 or 378'-0	Diesel Driven Fire Pump	Pump Room Floor or Intake Structure Roof	Pump Room Floor	B
		366'-0 or 378'-0	Jockey Pump Fire Pump	Pump Room Floor or Intake Structure Roof	Fire Pump	B
Intake Structure Gantry Crane (L7)	Intake Structure Unit 2	366'-0 or 378'-0	Unit 2 Service Water Pump Motor	Service Water Pump Intake Room Floor or Intake Structure Roof	Pump Room	B
		322'-6	Unit 2 Service Water Pump	Intake Structure Base Mat	None	B
Intake Structure Gantry Crane (L7)	Intake Structure Units 1 and 2	378'-0	Room Hatch Plugs	Intake Structure Roof	Pump Room	B

Appendix A (Cont.)

Crane	Location	Impact Elevation	Load	Impact Area	Safety-Related Equipment	Hazard Elim. Category*
Unit 1 Polar Crane (L2)	Reactor Bldg. Unit 1	424'-6	Missile Shield[a]	Top of Secondary Shield Walls	Reactor Vessel Head; RCPs "A" & "C"; Pressurizer; Steam Generators; Letdown Piping; Decay Heat Removal Piping	B, C
		376'-6	Top Head Insul. with w/Storage Racks[b]	Refueling Cavity Floor	Reactor Vessel Head	B
		362'-0	Transfer Tube Flange[b]	Refueling Cavity Floor	None	C
		376'-6 401'-6 or 357'-0	Reactor Vessel Head[b]	Refueling Cavity Floor; Head Stand; Equipment Hatch Area	Reactor Vessel; Decay Heat Removal Piping	B
		362'-0	Upper Plenum Assembly[b]	Reactor Vessel Refueling Cavity Floor	Reactor Vessel; Fuel in Vessel	B
		376'-6	Stud Storage Rack[b]	Refueling Cavity Floor	None	C
		376'-6	Indexing Fixture[b]	Refueling Cavity or Reactor Vessel Flange	Reactor Vessel	B
		376'-6	Head and Internal Handling Fixture Lift Rig w/Turn-buckles[b]	Reactor Vessel Head Refueling Cavity or Head Stand	Reactor Vessel Head; RCP "A"; "B" Steam Generator; Core Flood Piping	B

a. Loads with this designation can only be moved around the containment at elevation 424'-6" (426'-6" on Unit 2). These loads will be moved as close to the top of the secondary shield walls as possible with adjustments to the load's elevation as required to avoid such obstructions as handrails, Unit 2 main steam lines, etc.

b. This designation on the table indicates that these loads move over the reactor cavity.

Appendix A (Cont.)

Crane	Location	Impact Elevation	Load	Impact Area	Safety-Related Equipment	Hazard Elim. Category*
Unit 1 Polar Crane (L2) (Cont.)	Reactor Bldg. Unit 1	376'-6	Refueling Cavity Seal Plate[b]	Refueling Cavity or Reactor Vessel	RCP "A" or "B"; Reactor Vessel; "B" Steam Generator; Decay Heat Removal & Core Flood Piping	B
		357'-0 401'-6 or 424'-6	Unassembled ISI (ARIS) Tool	Equipment Hatch Area; Operating Floor South Cavity	"A" Cold Leg, "A" RCP, "B" Hot Leg; "B" Steam Generator; Core Flood Piping	B
		376'-6 424'-6	Assembled ISI (ARIS) Tool[b]	Top of "D" Ring, Reactor Vessel	"B" Hot Leg and Steam Generator; Reactor Vessel	B
		336'-6 or 357'-0	RCP Motor or Pump[a]	Bldg. Basement or Equipment Hatch	RCP; HPI Nozzles in Cold Leg; Reactor Bldg. Spray Header; Decay Heat Removal Piping and Core Flood Piping	B
		336'-6 or 357'-0	Structural Beams Above RCP "A", "D"[a]	Same as RCP Motor or Pump	Same as RCP Motor or Pump	B
		336'-6 or 357'-0	Structural Beams Above RCP "B", "C"[a]	Same as RCP Motor or Pump	Same as RCP Motor or Pump	B
		336'-6 357'-0 362'-0 376'-6 401'-6 or 424'-6	Fuel Transfer Carriage[b]	Bldg. Basement; Equipment Hatch; Operating Floor; Top of "D" Ring (South); Refueling Cavity Floor	HPI Nozzles "A" & "B"; Reactor Vessel; "B" Steam Generator	B

Appendix A (Cont.)

Crane	Location	Impact Elevation	Load	Impact Area	Safety-Related Equipment	Hazard Elim. Category*
Unit 1 Polar Crane (1L2) (Cont.)	Reactor Bldg. Unit 1	336'-6 357'-0 362'-0 376'-6 401'-6 or 424'-6	Upender[b]	Bldg. Basement; Equipment Hatch; Operating Floor; Top of "D" Ring (South); Refueling Cavity Floor	HPI Nozzles "A" & "B"; Reactor Vessel; "B" Steam Generator	B
		376'-0 357'-0 424'-6	Refueling Machine	Bldg. Basement; Equipment Hatch; Operating Floor; Top of "D" Ring (South); Refueling Cavity Floor	HPI Nozzles "A" & "B"; RCPs "A" & "B" Steam Generator	B
		376'-0 357'-0 424'-6	Auxiliary Refueling Machine	Bldg. Basement; Equipment Hatch; Operating Floor; Top of "D" Ring (South); Refueling Cavity Floor	HPI Nozzles "A" & "B"; RCPs "A" & "B" Steam Generator	B
		376'-0 357'-0 424'-6	Refueling Canal Ladder[b]	Bldg. Basement; Equipment Hatch; Operating Floor; Top of "D" Ring (South); Refueling Cavity Floor	HPI Nozzles "A" & "B"; RCPs "A" & "B" Steam Generator	B
		Any Elev.	Crane Load Block (Main Hoist)[b]	Any Area	Any Safety-Related Equipment Under a Load Path	D
Unit 2 Polar Crane (2L2)	Reactor Bldg. Unit 2	365'-0 426'-6	Head Maintenance Structure[b]	Refueling Cavity Floor; Top of Secondary Shield Walls	Reactor Vessel Head; Pressurizer; RCP "A"; RCP "C"; Safety Injection of Shutdown Cooling Piping	B

Appendix A (Cont.)

Crane	Location	Impact Elevation	Load	Impact Area	Safety-Related Equipment	Hazard Elim. Category*
Unit 2 Polar Crane (2L2) (Cont.)	Reactor Bldg. Unit 2	365'-0	Head	Refueling Cavity	Reactor Vessel	B
		357'-0 376'-0 426'-6	CEDM Ductwork[b]	Refueling Cavity Floor; Top of Secondary Shield Wall; Equipment Hatch	All RCPs; Both Steam Generators; Safety Injection Tanks "A" & "D"	B
		376'-0 426'-6	CEDM Cooling Shroud[b]	Refueling Cavity Floor; Top of Secondary Shield Wall	All the Above for CEDM Ductwork Plus "B" Hot Leg	B
		336'-6 426'-6 357'-0	RCP Motor and Pump[a]	Basement; Top of Secondary Shield Wall; Equipment Hatch Area	Respective RCP Pump; Pressurizer Surge Line; Safety Injection and Shutdown Cooling Piping; Cold Legs; Steam Generators	B
		336'-6 426'-6 357'-0	Structural Beams Above RCP "A" & "B"[a]	Basement; Top of Secondary Shield Wall; Equipment Hatch Area	Respective RCP Pump; Pressurizer Surge Line; Safety Injection and Shutdown Cooling Piping; Cold Legs; Steam Generators	B
		336'-6 426'-6 357'-0	Structural Beams Above RCP "C" & "D"[a]	Basement; Top of Secondary Shield Wall; Equipment Hatch Area	Respective RCP Pump; Pressurizer Surge Line; Safety Injection and Shutdown Cooling Piping; Cold Legs; Steam Generators	B
		336'-6 357'-0 376'-0 405'-6 426'-6	Inservice Inspection Tools[b]	Basement; Equipment Hatch Area; Refueling Cavity Floor; Operating Floor; Top of Secondary Shield Wall	RCP "C"; Reactor Vessel; Safety Injection and Shutdown Cooling Piping into "C" Cold Leg	B

Appendix A (Cont.)

Crane	Location	Impact Elevation	Load	Impact Area	Safety-Related Equipment	Hazard Elim. Category*
Unit 2 Polar Crane (2L2) (Cont.)	Reactor Bldg. Unit 2	354'-0 362'-0 405'-6 426'-6	Fuel Transfer Carriage[b]	Equipment Hatch Area; Refueling Cavity Floor; Operating Floor; Top of Secondary Shield Wall	"A" & "B" Cold Leg; "A" Hot Leg; Safety Injection and Shutdown Cooling Piping	B
		354'-0 362'-0 405'-6 426'-6	Upender[b]	Equipment Hatch Area; Refueling Cavity Floor; Operating Floor; Top of Secondary Shield Wall	"A" & "B" Cold Leg; "A" Hot Leg; Safety Injection and Shutdown Cooling Piping	B
		354'-0 362'-0 405'-6 426'-6	Refueling Machine	Equipment Hatch Area; Refueling Cavity Floor; Operating Floor; Top of Secondary Shield Wall	"A" & "B" Cold Leg; "A" Hot Leg; Safety Injection and Shutdown Cooling Piping	B
		376'-0 405'-6	Stud Tension and Hydraulic Unit[b]	Refueling Cavity Floor; Operating Floor	None	C
		357'-0 376'-0 365'-0 405'-6	Reactor Vessel Head[b]	Equipment Hatch Area; Refueling Cavity Floor; Operating Floor	Reactor Vessel Safety Injection and Shutdown Cooling Piping	B
		357'-0 376'-0 365'-0 405'-6	Reactor Vessel Head Lift Rig[b]	Equipment Hatch Area; Refueling Cavity Floor; Operating Floor	Reactor Vessel Safety Injection and Shutdown Cooling Piping	B
		405'-6	Reactor Vessel Head Studs	Operating Floor	None	C
		376'-0	Upper Guide Structure[b]	Refueling Cavity Floor	Reactor Vessel Internals; Fuel in Core	B
		376'-0	Upper Guide Structure Lift Rig[b]	Same as Upper Guide Structure	Same as Upper Guide Structure	B
		376'-0	Core Barrel[b]	Same as Upper Guide Structure	None	C

Appendix A (Cont.)

Crane	Location	Impact Elevation	Load	Impact Area	Safety-Related Equipment	Hazard Elim. Category*
Unit 2 Polar Crane (2L2) (Cont.)	Reactor Bldg. Unit 2	376'-0	Core Barrel Lift Rig[b]	Same as Upper Guide Structure	None	C
		336'-6 357'-0 405'-6 426'-6	Jib Crane (2L45)	Basement Floor; Head; Equipment Hatch Area; Operating Floor; Top of Secondary Shield Wall	Reactor Vessel; All RCPs; Both Steam Generators; Shutdown Cooling Piping	C
		376'-0 405'-6	Stud Stand w/Studs[b]	Refueling Cavity Floor; Operating Floor	None	C
		357'-0 376'-0 405'-6 426'-6	CEDM Extension Shaft Uncoupling[b]	Equipment Hatch Area; Refueling Cavity Floor; Operating Floor; Top of Secondary Shield Wall	Reactor Vessel Head; Pressurizer Surge Line to "A" Hot Leg; Shutdown Cooling Piping; "A" Cold Leg	B
		376'-0	Alignment Pins[b]	Refueling Cavity Floor; Vessel Flange	None	C
		376'-0 405'-6	Refueling Cavity Seal Plate[b]	Refueling Cavity Floor; Operating Floor	Reactor Vessel Head; Shutdown Cooling Pipe to "B" Hot Leg and "C" Cold Leg	B
		365'-0 405'-6 426'-6	Seal Plate Lift Rig[b]	Refueling Cavity Floor; Operating Floor; Top of Secondary Shield Wall	Same as Seal Plate	B
		Any Elev.	Main Hoist Load Block[b]	Any Area	Any Equipment Below Crane	D
MSIV Bridge Crane (2L10)	Turbine Auxiliary Bldg.	404'-0 386'-0 354'-0	Main Steam Isolation Valve (10,000 lbs)	Piping and Electrical Penetration Room Roof	Piping and Electrical Cables in North Penetration Rooms	B