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TECHNICAL EVALUATION REPORT

CONTROL OF HEAVY LOADS — PHASE II

TOLEDO EDISON COMPANY

DAVIS-BESSE NUCLEAR POWER STATION UNIT 1

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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.

Mr. I. H. Sargent, Mr. C. R. Bomberger, and Mr. R. J. Walter contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.

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 Davis-Besse Nuclear Power Station Unit 1. 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 the Toledo Edison Company (TEC), the Licensee for Davis-Besse Nuclear Power Station Unit 1, 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 August 9, 1983 [4]. On June 10, 1983, TEC 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 Davis-Besse Nuclear Power Station Unit 1. 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

2.2.1 Summary of Licensee Statements and Conclusions

Information provided by the Licensee identified the following load handling systems, capable of carrying 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 cask crane (140/20 tons)
2. In the vicinity of the reactor vessel
   - containment polar crane (180/25 tons)
   - reactor service crane (5 tons)
3. Over equipment required for safe shutdown
   - spent fuel cask crane
   - containment polar crane
   - component cooling water pump monorails
   - service water intake structure gantry crane.

2.2.2 Evaluation

The identification of the above load handling systems as being capable of carrying heavy loads within the reactor building is consistent with the intent of NUREG-0612.

In addition, however, the containment equipment jib cranes were also identified in References 4 and 7 as load handling systems which are capable of carrying heavy loads over the core, over spent fuel, or over safety-related equipment. The Licensee has not provided relevant information regarding these cranes in the Phase II response [5] to NUREG-0612.

2.2.3 Conclusion

The Licensee's statements and conclusions with respect to the load handling systems subject to the criteria of NUREG-0612 are consistent with those identified in Reference 5, with the exception of the containment equipment jib cranes. The containment equipment jib cranes were previously identified to be subject to the criteria of NUREG-0612; therefore, the Licensee should evaluate these cranes for compliance with NUREG-0612, Phase II criteria.

2.3 SPENT FUEL POOL AREA

2.3.1 Spent Fuel Cask Crane

2.3.1.1 Summary of Licensee Statements and Conclusions

The spent fuel cask crane is physically capable of carrying heavy loads over spent fuel in the spent fuel pool. Plant Technical Specification 3.9.7 provides various physical and administrative controls to prevent loads greater than 2,430 lb (weight of one fuel assembly and its handling tool) from being carried over spent fuel in the pool.

The spent fuel cask crane is electrically interlocked to prevent the crane from traveling over the spent fuel pool while any load is suspended from the main hook. The interlock can only be bypassed with a key obtained from the Shift Supervisor. Even upon bypassing, the main hook stays inoperative, and only the auxiliary hook can be used.

The spent fuel pool divider gates (4 tons) are the only loads handled in the immediate vicinity of spent fuel in the spent fuel pool. The Licensee analyzed the effects of an accidental load drop of the canal gates with respect to Criteria I through III of NUREG-0612, Section 5.1.

Criterion I

Assumptions used to determine the consequences of a load drop were based on the Davis-Besse FSAR and Safety Guide 25. Analyses performed by the Licensee indicated that the maximum number of fuel assemblies that could be damaged without exceeding NUREG-0612 guideline doses would be 15 assemblies. For the most limiting drop geometry of the divider gates, results of the Licensee's load drop analysis indicate that less than 15 fuel assemblies will be damaged, and therefore, the Licensee concludes that NUREG-0612, Criterion I is satisfied.

Criterion II

Regarding the criticality analysis, the Licensee stated that the assumptions of NUREG-0612, Section 2.2 are consistent with the existing design at the Davis-Besse plant. The fuel spacing of the fuel storage racks is designed to maintain K_{eff} at a value of less than 0.90. System Procedure SP 1104.42, "Spent Fuel Pool Operating Procedure," requires that the boron concentration in the spent fuel pool be maintained at a value greater than or equal to 1800 ppm. In addition, the License stated that the assumption was made that all fuel had an enrichment of 3.5 weight percent, which is the "highest probable enrichment" identified in the FSAR. Therefore, based upon a Davis-Besse design water-to-fuel ratio of 1.22, the Licensee stated that a load drop which causes crushing of the core will result in a decrease in K_{eff} and will not exceed 0.90, thereby satisfying Criterion II.

Criterion III

To determine the possibility of a load drop causing water leakage from the spent fuel pool, the Licensee performed a structural analysis of the potential for perforation or scabbing of the 5-ft spent fuel pool base. Results of this analysis indicate that perforation and scabbing are not probable. Penetration of the base is predicted to be insignificant and no leakage is anticipated, therefore satisfying this criterion.

2.3.1.2 Evaluation

Information provided by the Licensee identified two situations requiring evaluation of loads handled in the spent fuel pool area:

- o unrestricted movements of loads over the spent fuel pool precluded by electrical interlocks
- o bypassing the electrical interlocks to allow movement of the pool divider gates by the spent fuel cask crane auxiliary hook.

For routine movements of heavy loads outside the spent fuel pool, use of electrical interlocks to prevent load movements over the pool satisfies, to a large degree, the criteria of NUREG-0612, Section 5.1.2(2) for protection of spent fuel in the spent fuel pool. Additional information is required, however, to verify that adequate physical separation exists between the limits of crane travel (as restricted by electrical interlocks) and the walls of the spent fuel pool so that a dropped load (i.e., spent fuel cask) cannot tip or roll and damage the spent fuel pool wall or roll into the spent fuel pool and damage spent fuel.

The Licensee identified the movement of the pool divider gates using the auxiliary hook as the only instance in which the electrical interlocks will be bypassed. For this load movement, the Licensee performed analyses to demonstrate that the consequences of a load drop would not exceed Criteria I, II, and III of NUREG-0612, Section 5.1.

Criterion I

The Licensee assumptions appear consistent with those of NUREG-0612, Appendix A, and the analysis results indicate that the resultant fuel damage will not produce offsite radiological consequences which will exceed NUREG-0612 guidelines. Therefore, from information provided, it appears that Criterion I is satisfied for a load drop in the spent fuel pool area bounded by the weight of the pool divider gates.

Criterion II

Based upon comparison of the plant-specific design information provided by the Licensee with the information provided in Section 2.2 of NUREG-0612, it appears that K_{eff} following a load drop will not exceed a value of 0.90, which satisfies Criterion II. This conclusion is based upon the following information provided by the Licensee:

- o K_{eff} of spent fuel is less than 0.90 based upon design spacing of the spent fuel racks
- o spent fuel pool boron concentration is procedurally maintained greater than 1800 ppm
- o the "highest probable" fuel enrichment is 3.5 weight percent
- o design water/fuel ratio is 1.22.

However, order to fully agree that these criteria will satisfy Criterion II criticality concerns, additional assurance should be provided to substantiate the following concerns.

Although the Licensee stated that plant procedures exist which require that boron concentration be maintained greater than 1800 ppm in the spent fuel pool, insufficient information has been provided to ensure that the limit is enforced on a continuing basis by periodic sampling or to identify corrective actions to be taken in the event that concentration decreases to less than 1800 ppm. Similarly, although the Licensee assumes that all fuel impacted is enriched to less than 3.5 weight percent, additional information and assurances are needed to ensure that these "highest probable" FSAR values will not be exceeded.

Criterion III

Information provided by the Licensee appears to demonstrate adequately that no leakage will occur from the spent fuel pool as a result of a load drop of the pool divider gates, which satisfies this criterion of NUREG-0612.

2.3.1.3 Conclusion

Measures implemented by the Licensee in the vicinity of the spent fuel pool partially satisfy the criteria of NUREG-0612, Section 5.1.2(2). To fully satisfy the NUREG criteria, the following additional actions are required:

- For electrical interlocks which prevent load movement over the spent fuel pool, the Licensee should verify that adequate physical separation exists between the limits of crane travel and the spent fuel pool wall such that a dropped load will not impact, tip, or roll and cause damage to the spent fuel pool wall or spent fuel in the spent fuel pool.

- The Licensee should identify (1) the means of enforcing plant procedures on a continuing basis to ensure that boron concentration in the spent fuel pool does not decrease to less than 1800 ppm, as well as (2) the limitations imposed on load handling if this limit is violated. It is also requested that the Licensee provide a more definitive statement to ensure that the "highest probable" enrichment will not exceed 3.5 weight percent.

2.3.2 Reactor Service Crane

2.3.2.1 Summary of Licensee Statements and Conclusions

The reactor service crane is physically capable of carrying heavy loads over the reactor. At this time, the crane is not in use; therefore, the Licensee has deferred evaluation of this crane until it is put into service.

2.3.2.2 Evaluation and Conclusion

The operation of the reactor service crane cannot be independently evaluated until the Licensee provides an analysis of this crane with respect to the Phase II requirements of NUREG-0612.

2.4 REACTOR CONTAINMENT AREA

2.4.1 Containment Polar Crane

2.4.1.1 Summary of Licensee Statements and Conclusions

The containment polar crane is physically capable of carrying heavy loads over the reactor vessel. A load drop analysis for the postulated drops of the major loads carried by this crane is provided in Appendices A and B of

Reference 5. The Licensee does not consider it feasible to postulate a random mechanical failure of the crane load-bearing components when moving either the main hoist or auxiliary hoist load block without a load and has not included this load in the load drop analysis. However, the Licensee identified two possible failure modes that could result in the load drop of the main hook and load block:

1. A control system or operator error resulting in hoisting of the block to a "two blocking" position with continued hoisting by the motor and subsequent parting of the rope (this situation can be prevented by operator action prior to "two blocking" or by an upper limit switch to terminate hoisting prior to "two blocking").

2. Uncontrolled lowering of the load block due to failure of the holding brake to function (the likelihood of this can be made small by use of redundant holding brakes).

To prevent the occurrence of these load drops, the polar crane main and auxiliary hoists are provided with upper and lower limit switches along with a Revere digital weight indicator and limiter. The Revere digital weight indicator and limiter significantly reduces the likelihood of damage to the crane or lifting devices due to an overload, and the limit switches reduce the likelihood of two blocking. The main and auxiliary hoists are also equipped with dual electric brake systems that reduce the likelihood of uncontrolled lowering.

Therefore, the Licensee concluded that a drop of the load block and hook is of sufficiently low likelihood that it does not require a load drop analysis.

To demonstrate compliance with the requirements of NUREG-0612 for remaining heavy loads, the Licensee has performed both structural and systems analyses to demonstrate that the consequences of a load drop will not exceed the criteria of NUREG-0612, Section 5.1.2. In performing these evaluations, applicable heavy loads were identified, realistic load drop scenarios developed based on procedures in effect, and systems evaluations used to augment structural analyses which indicated local failure only. The detailed structural analyses that were performed assessed the structural response and consequences of a dropped load. Structural evaluation methods and criteria follow the criteria of NUREG-0612 and the recommendations of the ASCE technical committee on impulse and impact loads. Based upon conclusions contained in the following paragraphs, the Licensee concludes that NUREG-0612 Criteria I through III are met for all postulated load drops over the reactor vessel.

The structural evaluations were divided into two categories: those causing local failure only (spalling or scabbing) and those for which overall structural deformation or failure is anticipated. Detailed evaluations were performed only for those drops whose consequences were unacceptable.

The Licensee's analysis included an evaluation of a load drop of the plenum assembly (119,000 lb) into the core from the highest elevation possible (73.5 in), as dictated by physical restraints. (This height was determined by the height of the internals index fixture, which is used to ensure exact alignment of the assembly as it is removed and replaced.) The Licensee stated that the maximum kinetic energy associated with such a drop is 729,000 ft-lb, assuming that the load reaches the core and impact is uniformly transmitted to all fuel cells. The evaluation indicates that the strain in the individual fuel cells beyond buckling does not exceed an allowable strain level of 0.01, based upon the properties of the cladding. Therefore, it is the Licensee's conclusion that the fuel cladding will not rupture and no radioactive gases will be released.

A load drop of the reactor vessel head (RVH) has also been evaluated by the Licensee. The RVH (330,000 lb) has been postulated to drop from a height of 5 ft (based upon procedural limitations), and the impact is transmitted from the RPV flange through the RPV shell to the cold leg nozzles, which are the RPV ultimate supports. Evaluation indicates that the ultimate load that can be absorbed by these supports is 7 million pounds, whereas the maximum kinetic energy that is produced by the load drop is only 1.65 million pounds and is therefore acceptable based upon accident assumptions. The leak tight integrity of the reactor coolant pressure is also demonstrated.

An evaluation of a load drop of one of the six missile shields (94,500 lb) onto the control rod drive service structure has also been conducted. Procedures are in place which require the removal and replacement of these shields in specific sequential order so as to minimize the potential danger to the components located below. The Licensee stated that these procedures ensure that a dropped shield will fall back into place or will fall onto another shield located below. Therefore, based upon these considerations and the bounding load drop of the RVH, the consequences of a drop of a missile shield is acceptable.

In addition to the structural evaluations, the Licensee has also performed systems evaluations to demonstrate that acceptable consequences are present for specific cases. Several load drop scenarios have been defined, based upon the status of the RVH (installed or removed) and the size of the resulting unisolable RCS leak. Load drops reviewed in the reactor vessel area indicate that possible targets are the RCS nozzles and the core flood nozzles. Results of the structural analysis indicate that the RCS nozzles will remain intact. Based upon the assumption that only one core flood nozzle will rupture, the Licensee states that all cases that were evaluated indicate that adequate core cooling will be maintained.

2.4.1.2 Evaluation

The Licensee's analysis of the containment polar crane has been evaluated in accordance with the criteria of NUREG-0612, Section 5.1.3, for loads handled in the vicinity of the reactor vessel. The Licensee has identified crane design features which make the likelihood of damage from an unloaded load block extremely remote and has performed detailed analyses in an attempt to demonstrate that Criteria I, II, and III of NUREG-0612 have been satisfied.

It is agreed that for an unloaded load block, suitable design features in the form of upper and lower limit switches, dual electric brake systems, use of a weight indicator and limiter, and the large material safety margins associated with the lift of an unladen block eliminate the need for further analysis or crane modification.

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For other loads carried in the vicinity of the reactor vessel, the following are evaluations of the Licensee's statements with respect to the criteria of NUREG-0612, Section 5.1.

Criterion I

Analysis of information provided by the Licensee appears to adequately demonstrate that fuel will not be crushed and no adverse radiological consequences will be experienced, based upon the Licensee's conclusion that a drop of the 119,000-lb plenum assembly from a height of 73.5 inches is the limiting load drop condition. However, it is requested that the Licensee confirm that no other situations exist in which a physically smaller load may be dropped from a higher height and cause sufficient local damage to exceed the offsite radiological consequences of Criterion I. If the Licensee confirms that a drop of the plenum assembly bounds all other such load drops, then Licensee assumptions are consistent with NUREG-0612 and the Licensee will satisfy Criterion I in the vicinity of the reactor vessel.

Criterion II

Drops of the following loads which may cause criticality conditions have been analyzed by the Licensee: plenum assembly and missile shields. However, neither analysis appears to specifically address the issue of fuel crushing and the resulting potential for criticality. Therefore, the Licensee should provide additional information to confirm that these analyses demonstrate that criticality conditions will not be exceeded or perform appropriate analyses to address this issue.

Criterion III

The Licensee has performed structural evaluations of two load drops in the vicinity of the reactor vessel, as well as a systems analysis of this area, to demonstrate that Criterion III will not be violated. The two load drops are the drop of the reactor vessel head (RVH) from a height of 5 ft and a drop of a missile shield over the core. Unlike the plenum assembly, a drop of either load onto the reactor vessel does not require precision alignment with an indexing fixture. The RVH drop height of 5 ft is based upon procedural limitations only, while the missile shield evaluation is also predicated on procedural controls in which the shields are moved laterally and sequentially outward from the center of the reactor vessel at the minimum height necessary to clear other in-place shields. Orientation of the shields is not predicted by the Licensee to change, and therefore a drop will impact either another missile shield or the D-ring wall.

Evaluation of initial conditions for both a RVH drop or missile shield drop indicates that both analyses rely on procedural restrictions which limit the consequences of the load drop and therefore do not conservatively predict worst-case consequencs associated with a load drop. These assumptions are not consistent with Appendix A of NUREG-0612, which states that analyses should assume worst-case orientation and a drop from the maximum height in determining the consequences of a load drop. In general, such procedural controls are not equivalent to physical restraint, enhanced load handling reliability, or load drop analysis in demonstrating that load-drop consequences are acceptable or that the likelihood of a load drop is reduced. Therefore, structural analyses performed for the RVH and procedures controlling movements of the missile shields are not acceptable as a basis for compliance with Phase II.

In addition to structural evaluations, the Licensee has performed systems analyses to verify that core cooling will not be lost. Although the Licensee indicates that core cooling will not be lost, the following assumptions should be verified to be accurate:

1. The systems analysis is based upon results of the structural analysis in assuming that vessel integrity is not lost; since it has been requested that the Licensee reconfirm the validity of structural evaluations for worst-case consequences, acceptability of the systems approach is contingent upon this reevaluation.

2. No rationale has been provided to agree with the Licensee's assumption that only one of two core flood nozzles will be ruptured.

2.4.1.3 Conclusions

Although the Licensee has performed analyses to demonstrate compliance with the applicable criteria of NUREG-0612 for heavy loads handled in the vicinity of the reactor vessel, evaluation of these analyses indicates that assumptions used by the Licensee have not been clearly addressed or are not in accordance with the guidance of NUREG-0612, Appendix A. Therefore, Licensee action is requested to address the remaining concerns for each of the following criteria identified as follows.

Criterion I

The Licensee should verify that a drop of the plenum assembly presents the bounding load drop for analyzing damage to the fuel and resultant off-site consequences.

Criterion II

The Licensee should address the potential for exceeding limits for criticality specified in this criterion.

Criterion III

The Licensee should reanalyze the consequences of a load drop and its effects on structural integrity, using assumptions consistent with NUREG-0612, Appendix A, as well as the consequences of new structural analyses on the systems analysis. Use of procedural controls alone is not an adequate alternative to performance of analyses or improvement of load handling reliability to document conformance with Phase II guidelines.

2.5 OVERHEAD HANDLING SYSTEMS IN AREAS CONTAINING SAFE SHUTDOWN EQUIPMENT

2.5.1 Containment Polar Crane

2.5.1.1 Summary of Licensee Statements and Conclusions

The Licensee identified the containment polar crane to be capable of handling heavy loads over equipment required for safe shutdown. To demonstrate compliance with Criterion IV of NUREG-0612, Section 5.1, the Licensee performed detailed structural and systems analyses to demonstrate that core cooling can be maintained following any credible load drop. The containment was segmented into 9 regions for purposes of these analyses. In each of these areas, the License performed specific structural analyses based upon the limiting load drop which may occur in the region and systems analyses based upon the safe shutdown systems present within the region. Results of the integrated analyses are tabulated as follows:

Region 2: North End of Operating Deck

The limiting load drop is a drop of the reactor vessel head from a height of 7 ft. Structural analysis indicated structural deformations, and therefore all systems were assumed lost in the area. However, systems analysis indicated that decay heat removal would remain available and core cooling could be maintained.

Region 3: D-Ring Enclosure

The limiting load drop is drop of either a missile shield or a reactor coolant pump (RCP) motor into the steam generator area. Due to the complexity of the structural analysis for a drop of the RCP motor, the analysis was not performed, and it was assumed that the decay heat removal system components were lost. However, the Licensee stated that the requisite emergency core cooling system (ECCS) components would remain available to maintain core cooling, or a suitable arrangement could be made by aligning the PORV system.

Region 4 Refueling Canal

The limiting load drop is a drop of the missile shield from a height of 76 ft. Because the consequences of a drop of the missile shields indicated excessive structural deformations, a systems analysis was performed. It was assumed that a LOCA would occur in one loop and that one train of core flooding would be damaged. Results indicated that core cooling capabilities would not be lost and flooding and leakage level were acceptable with respect to survivability of equipment.

Region 5: South End of Refueling Canal

The limiting load drop is a drop of the plenum assembly on the cavity floor. Structural and systems analyses indicated that the consequences of such a drop would be insignificant and that load movements within this area are acceptable.

Region 6: Piping Enclosures

Impact due to secondary missiles are the only potential problems in this area because physical interference prevents movements of heavy loads over this area. Systems analysis assumed that one line of decay heat removal would be lost; however, analysis results indicated that adequate redundancy would exist to maintain core cooling.

Region 7: SE Quadrant Grating

As in region 6, no heavy loads may be moved over this region. Assuming loss of decay heat removal suction piping in this area, analysis indicated that the ECCS would still be available and core cooling would be maintained.

Region 8: In-Core Instrument Area

The limiting load drop is a drop of the in-core instrument tank access hatch cover (3.4 tons). Structural analysis indicated that a drop of this load will not perforate the 606-ft elevation slab. Systems analysis assumed that all instrument tubes located in the trunk would be severed, causing a LOCA. Analysis results indicated that the ECCS would still be available and core cooling would be maintained.

Region 9: Area Adjacent to Equipment Hatch

The limiting load drop is a drop of a RCP motor. No structural analysis was performed. Systems anlysis assumed loss of makeup and purfication piping, resulting in a LOCA. Analysis results indicated the ECCS would remain unaffected and core cooling would be maintained.

2.5.1.2 Evaluation

The analyses performed by the Licensee to evaluate the consequences of load drops onto equipment required for safe shutdown appear to satisfy, to a large degree, the evaluation criteria of NUREG-0612, Criterion IV. The Licensee determined, through a combination of structural and systems analyses, that the consequences of a load drop will not preclude the ability to maintain core cooling. Licensee assumptions are the same as those previously identified in Section 2.4.1 of this evaluation and are consistent, to a large degree, with the guidance of NUREG-0612.

It is noted that Appendix A of NUREG-0612 specifies that load analyses should consider a load drop from the maximum height at any point within the unrestrained movement of the crane. In determining the limiting load drop for each region identified within the Davis-Besse containment, however, the Licensee appears to have placed significant reliance on administrative controls which direct the movements of loads, such that certain load movements will not be conducted in certain areas of the containment. In addition, credit appears to have been taken for other restrictions (i.e., lift height) to reduce the effects of certain load drops. The Licensee appears to rely on the use of administrative controls to eliminate from further consideration certain heavy loads handled in the vicinity of safe shutdown equipment. 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. 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.

2.5.1.3 Conclusion

Analyses performed by the Licensee partially demonstrate that core cooling can be maintained following a load drop in the containment onto equipment required for safe shutdown. However, additional information is needed from the Licensee to identify and justify the administrative controls that are used to restrict movements of loads within the various regions, and for which credit appears to have been taken in both the selection of the limiting load drop and to mitigate potential consequences of various load drops.

2.5.2 Component Cooling Pump Monorails

2.5.2.1 Summary of Licensee Statements and Conclusions

The limiting load drop for the component cooling water pump monorails is a drop of the component cooling water pump. Based on results of the structural analysis, the Licensee concluded that "perforation and scabbing were not probable" and any possible "effects were found to be insignificant." Systems evaluation indicated that adequate physical separation exists so that suitable system redundancy would be retained and safe shutdown functions would not be lost.

2.5.2.2 Evaluation and Conclusion

Analyses of a load drop by the component cooling water pump monorails indicated that existing design is adequate to satisfy Criterion IV of NUREG-0612, Section 5.1. Assumptions used by the Licensee were generally consistent with those identified in NUREG-0612, Appendix A.

2.5.3 Intake Structure Gantry Crane

2.5.3.1 Summary of Licensee Statements and Conclusions

The limiting load drop in the service water intake structure area is a drop of the service water pump motors. Results of the structural analysis indicated that "perforation or scabbing were not probable," although it was considered in the systems evaluation. Systems analysis indicated that adequate physical separation exists between system components so that system redundancy would not be jeopardized and system functions would remain operable.

2.5.3.2 Evaluation and Conclusion

Assumptions used by the Licensee to perform structural and systems analysis appear to be generally consistent with the intent of NUREG-0612. Based upon analysis results, Criterion IV of NUREG-0612, Section 5.1 is satisfied.

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 that should be addressed in seeking to resolve issues where the degree of load handling reliability provided by cranes at Davis-Besse Nuclear Power Station Unit 1 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 information provided by the Licensee has been assessed as insufficient to support an independent conclusion that load handling reliability is consistent with the evaluation criteria of Section 2.1 in the following areas:

Load Handling System Evaluations (Sections 2.2.2 and 2.3.2.2)

> The Licensee should evaluate the containment equipment jib cranes for compliance with the criteria of NUREG-0612. In addition, an evaluation of the reactor service crane must be deferred since the crane is not in use and has not been evaluated by the Licensee.

Spent Fuel Pool Interlocks (Section 2.3.1.3)

> The Licensee should verify that adequate physical separation exists between the limits of crane travel and the spent fuel pool wall so that a dropped load will not impact, tip, or roll and cause damage to the spent fuel pool wall or spent fuel in the spent fuel pool.

Movements Within the Spent Fuel Pool (Section 2.3.1.3)

> To allow movements of the pool divider gates, the Licensee should identify the means of enforcing plant procedures on a continuing basis to ensure that boron concentration in the spent fuel does not decrease to less than 1800 ppm, as well as identify load handling limitations if this limit is

violated. In addition, the Licensee should provide necessary assurances that maximum enrichment will not exceed 3.5 weight percent.

Load Drops in the Vicinity of the Reactor Vessel (Section 2.4.1.3)

To fully satisfy Criterion I, the Licensee should verify that a drop of the plenum assembly presents the bounding load drop for analyzing damage to the fuel and resultant offsite consequences.

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 Criticality Analysis (Section 2.4.1.3)

To demonstrate compliance with Criterion II, the License should address the potential for exceeding limits for criticality specified in this criterion. Demonstration that a drop of a missile shield is not probable due to procedural controls is not adequate justification that Criterion II will not be exceeded.

Reactor Vessel Area Structural Integrity (Section 2.4.1.3)

The Licensee should reanalyze the consequences of a load drop onto the reactor vessel and its effects on structural integrity, using assumptions consistent with NUREG-0612, Appendix A, as well as reevaluate the consequences of new structural analyses on the systems analysis.

Load Drops onto Components Required for Safe Shutdown (Section 2.5.1.3)

The Licensee appears to rely on the use of administrative controls to eliminate from further consideration certain heavy loads which are handled in various regions of the containment. 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 cricumstances (specifically where the administrative controls provide large separations between the control limits and the impact area of interest which 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

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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 Davis-Besse Nuclear power Station Unit 1"
   TER-C5506-348, August 9, 1983

5. R. P. Crouse (TEC)
   Letter to J. F. Stolz (NRC)
   Subject: Control of Heavy Loads (Phase II)
   June 10, 1983

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

7. R. P. Crouse (TEC)
   Subject: Control of Heavy Loads (Phase I)
   February 1, 1982

8. FRC
   Draft Technical Evaluation Report, "Control of Heavy Loads at Davis-Besse Nuclear Power Station Unit 1"
   TER-C5506-530, August 27, 1983

9. Code of Federal Regulations
   Energy (10CFR100)
   10 - (Parts 0 to 199)
   January 1, 1983

10. Updated FSAR
    Davis-Besse Nuclear Power Station Unit 1
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