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EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards addl info re control of heavy loads in response to 801222 request.Heavy loads handled w/reactor bldg crane main hoist & lifting devices.Load lift points required by NUREG-0612 provided,

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TITLE: Control of Heavy Loads Near Spent Fuel (USI A-36) Operating Reactor

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

August 1, 1982

Director of Licensing
Attention: Mr. Darrell G. Eisenhut
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Eisenhut:

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Your December 22, 1980 letter requested information regarding the control of heavy loads at nuclear power plants. Our July 28, 1981 letter provided the information requested in Section 2.1 of Enclosure 3 to your letter. The attachment to this letter provides the remainder of the information requested in Sections 2.2 and 2.3 of Enclosure 3 to your December 22, 1980 letter.

Sincerely,

NIAGARA MOHAWK POWER CORPORATION

T E Lempges

T. E. Lempges
Vice President, Nuclear Generation

BDW/jab
Attachments

A033

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Niagara Mohawk Power Corporation
Nine Mile Point Unit 1
Docket 50-220
DPR-63

Control of Heavy Loads

August 1, 1982

Provided herein is the information requested in Sections 2.2 and 2.3 of Enclosure 3 to the Nuclear Regulatory Commission's letter of December 22, 1980. This information supplements and, where appropriate, supersedes information provided in our previous submittals.

SECTION 2.2 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN THE REACTOR BUILDING

NUREG-0612, Section 5.14 provides guidelines concerning the design and operation of load handling systems in the vicinity of spent fuel in the reactor vessel or in storage. Information provided in response to this section should demonstrate that adequate measures have been taken to ensure that in this area either the likelihood of a load drop which might damage spent fuel is extremely small or that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG-0612 Section 5.14, Criteria I through III.

ITEM 2.2.1 Identify by name, type, capacity and equipment designator any cranes physically capable (i.g. ignoring interlocks, movable mechanical stops or operating procedures) of carrying loads over spent fuel in the storage pool or in the reactor vessel.

RESPONSE: The Reactor Building crane is capable of carrying loads over nearly all locations at elevation 340 feet in the Reactor Building, including over the reactor vessel and the spent fuel pool. The Reactor Building crane is an overhead bridge crane with a main hoist of 125 ton capacity and an auxiliary hoist of 25 ton capacity. A one-half ton overhead hoist is also provided.

Additionally, the refueling bridge is capable of carrying loads over the reactor vessel and the spent fuel pool. However, this crane is used for transfer of fuel, it is not used for handling heavy loads (a heavy load is defined as anything weighing more than a fuel assembly, or approximately 700 pounds).

The jib crane can be mounted in several positions around the spent fuel pool and the reactor cavity. It can be used for assisting refueling operations by handling small tools, channel assemblies, or LPRM's. However, the jib crane is not used for handling heavy loads at Nine Mile Point Unit 1.

ITEM 2.2.2 Justify the exclusion of any cranes in this area from the above category by verifying that they are incapable of carrying heavy loads or are permanently prevented from movement of heavy loads over stored fuel or into any location where following any failure such load may drop onto the reactor vessel or spent fuel storage pool.

RESPONSE: The refueling bridge crane is used for handling of fuel during refueling operations but it is not used for the handling of heavy loads. On this basis, this crane may be excluded from the criteria of NUREG-0612.

Additionally, the jib crane and the one-half ton capacity overhead hoist may similarly be excluded since they are not used to handle heavy loads.

ITEM 2.2.3 Identify any cranes listed in 2.2-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG-0612, Section 5.1.6 or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE: The reactor building crane trolley was replaced with a new trolley that included a main hoist system having dual load path capability which substantially complies with Section 5.1.6 of NUREG-0612. The adequacy of this handling system to satisfy NRC criteria for redundancy and load handling reliability characteristics has previously been reviewed and accepted by the staff. By letter of July 26, 1973 from Mr. P. D. Raymond to Mr. Donald J. Skovholt, a description was provided of the redundant hoisting system which has been installed at Nine Mile Point Unit 1. Additional information related to this redundant system was provided by letter of December 10, 1975 from Mr. G. K. Rhode to Mr. K. R. Goller. Subsequent correspondence and meetings between Niagara Mohawk and the staff resolved questions raised by the NRC in their evaluation of this redundant hoisting system. The hoisting system was approved by letter of January 7, 1976 from Mr. George Lear to Mr. G. K. Rhode. On this basis we believe that the Nine Mile Point Unit 1 reactor building crane main hoist is provided with sufficient design features to make the likelihood of a load drop due to failure of main hoist components to be extremely small for loads carried by the main hoist.

Additionally, evaluations have been performed of the design adequacy of lifting devices and interfacing lift points used for handling of heavy loads with the reactor building crane main hoist. Table A-1 lists the heavy loads that are handled by the reactor building crane main hoist and identifies the associated lifting device that is used for the handling of that load. The design evaluations considered the maximum static plus dynamic loads that these lifting devices would be subjected. NUREG-0612, Section 5.1.6, requires that lifting devices and interfacing lift points satisfy ANSI N14.6-1978, Section 6 for the maximum static plus dynamic load. Calculations were performed to determine the maximum dynamic loading that could be imparted on lifting devices. These calculations conservatively used: no load on the crane to determine maximum deceleration; lowering of the hook at 110% of hoist speed when both holding brakes are suddenly applied; rotational inertia of only the motor (ignoring gearing, drum, and brake wheel inertia); and operation of the brakes at full rated torque. These calculations determined that maximum deceleration of the load due to the load brakes would be less than 3% of the load due to gravity. With the rated load, the deceleration load would be smaller. Based on these calculations, a conservative factor of 5% was used for dynamic loads on lifting devices.

Based on these design evaluations it was found that a number of the lifting devices and interfacing lift points for main hoist heavy loads satisfied the criteria of Section 5.1.6. Table A-2 provides a listing of the lifting devices and the interfacing lift points that were found to be in compliance with Section 5.1.6 of NUREG-0612. However, certain lifting devices and interfacing lift points did not satisfy these criteria. The lifting devices and lift points that do not meet the criteria of Section 5.1.6 are listed in Table A-3 along with an identification of components that do not meet the criteria. Table A-3 also indicates the design safety factors that are achieved for these components.

Although the devices identified in Table A-3 do not comply with the recommendations of Section 5.1.6 of NUREG-0612 as it pertains to safety factors, these devices do utilize safety factors which are essentially in compliance with normal industry practice. It is Niagara Mohawk's judgment that the lifting devices identified in Table A-3, when used in conjunction with inspection and maintenance procedures and crane operator training provide sufficient assurance that a load drop will not occur. The basis for this judgment is as follows:

Safety factors normally used throughout industry provide assurance that lifting devices subject to frequent use, harsh environments, and infrequent inspections will function properly. The lifting devices identified in Table A-3 are used on a very infrequent basis, are not subject to harsh environments, and will be inspected prior to use.

As noted in Section 4 of NUREG-0612, the large majority of crane accidents are the result of crane operator error or improper rigging. Niagara Mohawk has in place a lesson guide which is used to train crane operators. This lesson guide provides assurance of proper crane operation. To reduce the likelihood of improper rigging, Niagara Mohawk will review lifting procedures used at Nine Mile Point Unit 1 and revise them as necessary to specify lifting devices to be used with the various loads. Methods will be developed to ensure positive identification of lifting devices. Additionally, procedures will be revised to require a visual inspection of the lifting devices prior to use.

Based on the features of the Reactor Building crane main hoist and the steps being taken to implement inspection and identification of lifting devices, Niagara Mohawk believes that the likelihood of a load drop associated with the main hoist is extremely remote.

TABLE A-1

HEAVY LOADS HANDLED WITH THE REACTOR BUILDING

CRANE MAIN HOIST

LOAD	LIFTING DEVICE
Reactor Vessel Head	Vessel head lifting rig and 3 - "S7" slings
Reactor Cavity Shield Plugs	2 - "S1" slings
Drywell Head	Drywell head lifting assembly
Steam Dryer	Dryer/Separator sling assembly
Moisture Separator and Shroud Cover	Dryer/Separator sling assembly
Large Canal Pool Plugs	2 - "LB-1" brackets with spreader beam and 2 - "S5A" slings (or) Underwater lifting rig and 2 - "S5A" slings
Shield Platform	Reactor head lifting device with the shield platform strongback
Shield Support Structure	Shield platform strongback with turnbuckles
Fuel and Equipment Casks	Cask yokes

TABLE A-2

LIFTING DEVICES AND LOAD LIFT POINTS

THAT SATISFY NUREG-0612 (5.1.6)

LIFTING DEVICES

Vessel Head Lifting Rig
Vandenburgh Cask Yoke Adapter

LOAD LIFT POINTS

Vessel Head Lifting Lugs
Drywell Head Lifting Lugs
Steam Dryer Lifting Lugs
Moisture Separator Lifting Lugs
Shield Platform Lift Points
Curtain Shield Support Structure Lifting Lugs
Cavity Shield Plug Lifting Lugs

TABLE A-3

LIFTING DEVICES AND LOAD LIFT POINTS

THAT DO NOT MEET NUREG-0612 (5.1.6)

LIFTING DEVICE/ LIFT POINT	COMPONENT NOT MEETING CRITERIA	SAFETY FACTORS ACHIEVED YIELD	ULTIMATE
Sling "S1"	Pear Link	-	4.85
	Wire Rope	-	5.49
	Tapered Sleeves	-	5.07
	Turnbuckles	-	5.19
	Anchor Shackles	-	7.61
Sling "S5A"	Pear Link	-	5.93
	Wire Rope	-	6.01
	Torpedo Look Locks	-	6.56
	Turnbuckles	-	6.66
Sling "S7"	Wire Rope	-	6.33
	Tapered Sleeves	-	5.86
	Turnbuckles	-	5.86
Drywell Head Lifting Assembly	Wire Rope Link	-	6.4
	Anchor Shackle	-	4.8
	Lifting Arms (Bending)	3.6	4.0
Bracket "LB-1"	Lifting Lug (Bending)	4.0	6.5
	Weld (Shear)	-	7.2
	Bottom Plate (Bending)	3.0	4.9
Shield Platform Strongback	Lifting Arms (Bending)	6.32	7.37
	Turnbuckles	-	9.05
Dryer/Separator Sling Assembly	Wire Rope	-	9.4
	Forged Ring	-	5.95
	Lifting Plates (Shear)	5.1	8.2
	Clevis and Rod	-	5.6
Underwater Lifting Rig	Bolts B1 and B2	4.6	7.3
Large Canal Plug Lift Points	Anchor Rods (Tension)	4.2	6.0

ITEM 2.2.4 For cranes identified in 2.2-1, above, not categorized according to 2.2-3, demonstrate that the criteria of NUREG-0612, Section 5.1, are satisfied. Compliance with Criterion IV will be demonstrated in response to Section 2.4 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the Reactor Building and your determination of compliance. This response should include the following information for each crane:

a. Where reliance is placed on the installation and use of electrical interlocks or mechanical stops, indicate the circumstances under which these protective devices can be removed or bypassed and the administrative procedures invoked to ensure proper authorization of such action. Discuss any related or proposed technical specifications concerning the bypass of such interlocks.

b. Where reliance is placed on the operation of the Standby Gas Treatment System, discuss present and/or proposed technical specifications and administrative or physical controls provided to ensure that these assumptions remain valid.

c. Where reliance is placed on other site-specific considerations (e.g., refueling sequencing), provide present or proposed technical specifications, and discuss administrative or physical controls provided to ensure the validity of such considerations.

d. Analyses performed to demonstrate compliance with Criteria I through III should conform to the guidelines of NUREG-0612, Appendix A. Justify any exception taken to these guidelines, and provide the specific information requested in Attachment 2, 3, or 4, as appropriate, for each analysis performed.

RESPONSE: The reactor building crane auxiliary hoist handles a number of heavy loads as noted in Table A-4. Potential impact areas of these heavy loads are identified in Table A-5. Table A-6 identifies the potential safety concerns, and Table A-7 summarizes the various load-impact regions considered, evaluations performed, and results obtained. Attachment B provides a description of the methodology used for the load drop evaluations that were performed.

Response 2.3.2 discusses the evaluations performed where heavy loads could be dropped in Regions 3-6; the following discusses the potential effects due to load drops in Regions 1 and 2.

Reactor Vessel and Reactor Cavity (Region 1)

Several heavy loads of less than 10 tons are handled by the auxiliary hoist over or in close proximity to the reactor vessel or vessel cavity. The following load-impact combinations were identified as limiting cases for auxiliary hoist load drops in this region:

1. Drop of a small canal plug that tips and falls into the cavity, damaging the drywell head and impacting the vessel head, and resulting in loading of the vessel skirt;

2. Drop of the reactor head insulation assembly onto the vessel head, resulting in loading of the vessel skirt; and

3. Drop of the portable radiation shield ("Japanese Bridge") onto the separator assembly in the vessel.

The analysis of the first and second load-impact combinations above considered the resulting loads in the reactor vessel skirt and the potential for excessive stresses.

The analysis of the third load-impact combination evaluated the potential for a drop of the portable radiation shield to lead to damage of the separator assembly such that fuel in the core could be impacted; and also evaluated the adequacy of core supports to withstand the loading that might result.

The result of all three of these analyses was that the evaluation criteria of Section 5.1 of NUREG-0612 are satisfied.

Spent Fuel Pool (Region 2)

The heavy loads that are handled by the reactor building crane auxiliary hoist are identified in Table A-4. These loads are not handled over irradiated fuel in the spent fuel pool. However, certain of these are handled in proximity to the spent fuel pool. Therefore, conservative analyses were performed to determine the potential consequences of a load drop. The heavy loads that were considered in evaluating potential consequences of load drops into the spent fuel pool were the small canal plugs, the spent fuel pool gates, and the jib crane. Structural analyses were performed to estimate the number of fuel rods that could be damaged as a result of a drop of either spent fuel pool gate, the jib crane, or any of the three small canal plugs onto the spent fuel in the storage racks. Based on this number of fuel rods (bounded by the drops of any small canal plug), dose calculations were performed for drops into the spent fuel pool using as a basis the information contained in the Nine Mile Point Unit 1 FSAR. However, the analysis in the FSAR was updated to conform with current criteria in the areas of fractions of fission products located in fuel plenums, filter efficiency assumptions, fumigation condition assumptions for elevated releases, and increased power level. These changes resulted in a worst-case complete accident period dose of 92.67 Rem to the thyroid and 7.93 Rem to the whole body, both at the site boundary, which was assumed for purposes of this analysis to be contiguous to the low population zone. Both doses exceed the 1/4 of 10 CFR Part 100 limit imposed by Criterion 1 of NUREG-0612, Section 5.1. However, they are both within the 10 CFR Part 100 limits of 300 Rem (thyroid) and 25 Rem (whole body). The worst case 2 hour doses, also calculated at the minimum site boundary distance, were 4.32 Rem (thyroid) and 0.81 Rem (whole body), both well within 10 CFR Part 100 limits.

Because the other loads (jib crane and both spent fuel gates) weigh much less than the plugs and thus impart less energy, fewer rods will be damaged, and the resulting doses would be well within NUREG-0612 criteria.

The potential for criticality as a result of a heavy load dropped on spent fuel stored in the spent fuel pool cannot be precluded. The spent fuel storage racks currently in use at Nine Mile Point Unit 1 rely on spacing to maintain k_{eff} less than .95. Crushing the fuel may result in an increase of k_{eff}.

The results of the preceding analysis indicate that complete compliance with the criteria of Section 5.1 of NUREG-0612 does not currently exist for loads handled by the auxiliary hoist. However, Niagara Mohawk believes that the use of the auxiliary hoist for handling loads, when supplemented with improved procedures, meets the intent of NUREG-0612 to provide a highly reliable load handling system, as discussed below.

All of the loads handled by the auxiliary hoist are well below the 25 ton capacity of that hoist. Based on this, a load drop due to a failure of the hoist mechanical components is considered an unlikely event. An evaluation will be performed to provide an assessment of the auxiliary hoist reliability.

The two most likely causes for a load drop when using the auxiliary hoist are two blocking events and an uncontrolled lowering of the load due to a failure of the holding brake to function. The Nine Mile Point Unit 1 crane auxiliary hoist is provided with two redundant and diverse upper limit switches to interrupt power to the hoist motor prior to "two blocking". One of the two limit switches is a geared limit switch driven off the drum shaft. The other is a counter weight switch that is released when the load block comes up against a trip bar. The holding brake has sufficient capacity to hold the rated load and is rated at 150% of full motor torque. The holding brake is solenoid released and spring applied on loss of power to the solenoid. Based on these features, it is unlikely that a load drop would occur as a result of "two blocking" or failure of the brake to function.

As stated previously, Niagara Mohawk believes that the likelihood of a load drop can be minimized if normal industry safety factors are used for lifting devices and improved inspection and identification procedures are implemented. Therefore, procedures addressing the handling of the small canal plugs, the jib crane, and the spent fuel gates will be revised to require a visual inspection and positive identification of lifting devices prior to use. Additionally, a review will be conducted to verify the adequacy of the lifting devices and lift points with regard to safety factors.

SECTION 2.3 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN PLANT AREAS CONTAINING EQUIPMENT REQUIRED FOR REACTOR SHUTDOWN, DECAY HEAT REMOVAL, OR SPENT FUEL POOL COOLING

NUREG-0612, Section 5.1.5, provides guidelines regarding the design and operation of load-handling systems in the vicinity of equipment or components required for safe reactor shutdown and decay heat removal. Information provided in response to this section should be sufficient to demonstrate that adequate measures have been taken to ensure that in these areas, either the likelihood of a load drop which might prevent safe reactor shutdown or prohibit continued decay heat removal is extremely small, or that damage to such equipment from load drops will be limited in order not to result in the loss of these safety-related functions. Cranes which must be evaluated in this section have been previously identified in your response to 2.1-1, and their loads in your response to 2.1-3-c.

ITEM 2.3.1 Identify any cranes listed in 2.1-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG-0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE: Our first report of July 28, 1981 excluded the handling systems other than the reactor building overhead crane from the criteria of NUREG-0612. The response to item 2.2.3 provides a discussion of the reactor building crane main hoist redundant load path features.

ITEM 2.3.2 For any cranes identified in 2.1-1 not designated as single-failure-proof in 2.3-1, a comprehensive hazard evaluation should be provided which includes the following information:

a. The presentation in a matrix format of all heavy loads and potential impact areas where damage might occur to safety-related equipment. Heavy loads identification should include designation and weight or cross-reference to information provided in 2.1-3-c. Impact areas should be identified by construction zones and elevations or by some other method such that the impact area can be located on the plant general arrangement drawings. Figure 1 provides a typical matrix.

b. For each interaction identified, indicate which of the load and impact area combinations can be eliminated because of separation and redundancy of safety-related equipment, mechanical stops and/or electrical interlocks, or other site-specific considerations. Elimination on the basis of the aforementioned consideration should be supplemented by the following specific information:

   (1) For load/target combinations eliminated because of separation and redundancy of safety-related equipment, discuss the basis for determining that load drops will not affect continued system operation (i.e., the ability of the system to perform its safety-related function).

   (2) Where mechanical stops or electrical interlocks are to be provided, present details showing the areas where crane travel will be prohibited. Additionally, provide a discussion concerning the procedures that are to be used for authorizing the bypassing of interlocks or removable stops, for verifying that interlocks are functional prior to crane use, and for verifying that interlocks are restored to operability after operations which require bypassing have been completed.

   (3) Where load/target combinations are eliminated on the basis of other, site-specific considerations (e.g., maintenance sequencing), provide present and/or proposed technical specifications and discuss administrative procedures or physical constraints invoked to ensure the validity of such considerations.

c. For interactions not eliminated by the analysis of 2.3-2-b, above, identify any handling systems for specific loads which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small and the basis for this evaluation (i.e., complete compliance with NUREG-0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

d. For interactions not eliminated in 2.3-2-b or 2.3-2-c, above, demonstrate using appropriate analysis that damage would not preclude operation of sufficient equipment to allow the system to perform its safety function following a load drop (NUREG-0612, Section 5.1, Criterion IV). For each analysis so conducted, the following information should be provided:

   (1) An indication of whether or not, for the specific load being investigated, the overhead crane-handling system is designed and constructed such that the hoisting system will retain its load in the event of seismic accelerations equivalent to those of a safe shutdown earthquake(SSE).

   (2) The basis for any exceptions taken to the analytical guidelines of NUREG-0612, Appendix A.

   (3) The information requested in Attachment 4.

RESPONSE: The reactor building crane auxiliary hoist handles a number of heavy loads as noted in Table A-4. Niagara Mohawk believes the auxiliary hoist provides a very reliable means of handling these loads, thereby meeting the intent of NUREG-0612. However, conservative analyses have been performed to assess the degree of compliance with criteria I through IV of NUREG-0612. Potential impact areas of these heavy loads are identified in Table A-5. Table A-6 identifies the potential safety concerns, and Table A-7 summarizes the various load-impact regions considered, evaluations performed, and results obtained. Response 2.2.4 discussed potential effects due to load drops in Regions 1 and 2. Attachment B describes the methodology that was used for performing the load drop evaluations. The following discusses evaluations performed where heavy loads could be dropped in Regions 3-6.

Reactor Internals Storage Pit (Region 3)

The reactor internals storage pit is used for storage of the dryer and the separator units during refueling/major maintenance, and for storage of the shield platform and the curtain shield support platform during plant operation. All of these heavy loads are handled with the reactor building crane main hoist. No heavy loads that are handled by the auxiliary hoist were identified that are routinely handled in this region. Accordingly, no load drop analyses were performed for this region.

Equipment Hatch (Region 4)

Certain loads are handled over the equipment hatch in the northwest corner of the reactor building. Loads that are handled in this equipment hatch may be hoisted as high as 90' above one of the corner rooms that contain certain safety related equipment. The largest load that is handled by the reactor building crane auxiliary hoist in this hatchway is a cluster of three new fuel containers handled at one time. A structural load drop analysis was performed for dropping a cluster of three fuel containers loaded with new fuel. Due to the large reinforced concrete slab for the railroad bay, the results of this analysis indicate that such a postulated load drop would not result in scabbing from below the railroad bay area, nor would it result in overall structural failure. Based on this analysis a load drop of new fuel in metal containers is within the criteria of Section 5.1 of NUREG-0612.

Platform Over Emergency Condensers (Region 5)

The emergency condensers are located on the operating deck, elevation 340' of the reactor building. A platform with steel grating is located over the top of the emergency condensers. No main hoist loads are carried over this region. In general, miscellaneous small loads are set down on this platform. The only heavy loads that were identified as being carried in this region are the dryer/separator sling assembly and the cask base plate that is attached to the bottom of shipping casks prior to lowering into the spent fuel pool cask drop protection system.

Load drop analyses were performed for these two heavy loads to determine the potential for perforation of the decking or overall structural failure of the platform.

As a result of these evaluations it was found that minor deformation of the decking may occur, but perforation or structural failure is not expected to occur, and emergency condenser functions will not be impaired.

Elevation 340 Foot Floor (Region 6)

Certain loads carried by the reactor building crane auxiliary hoist are carried over floor areas at elevation 340. Certain safety related equipment is located at elevations of the reactor building below this floor. Most of these loads are small such as new fuel in containers, the dryer/separator sling assembly, the jib crane, etc. The largest of these are the small canal plugs and the portable radiation shield ("Japanese Bridge"). However, the small canal plugs are the most limiting in terms of potential for damage. The evaluations of load drops of these heavy loads has indicated that a drop from the maximum carrying height that could reasonably be expected could result in scabbing of concrete from below the elevation 340' floor, but would not result in perforation or overall structural failure. As stated earlier, to resolve concerns associated with load drops, procedures requiring inspection and identification of lifting devices will be implemented.

TABLE A-4

HEAVY LOADS HANDLED WITH THE
REACTOR BUILDING CRANE AUXILIARY HOIST

LOAD	WEIGHT/TONS
Fuel Transfer Shield ("Japanese Bridge")	6.5
Small Canal Slot Plugs	9.5/9.75 1/
Fuel Pool Gates	0.5/0.65
Jib Crane	1.7
Reactor Head Insulation	5
Vessel Stud Rack	1-2
New Fuel Containers	1 2/
"Hydrolaser"	2
Fuel Channel Containers	0.9
Dryer/Separator Sling Assembly	1.5
Auxiliary Hoist Load Block and Hook	1.7
LPRM Shipping Box	0.94
Curtains for Texas Tower	1.68/1.64 3/

1/ The upper plug is 9.5 tons; the two lower plugs are 9.75 tons each.

2/ The weight shown is the loaded weight of a new fuel container. The new fuel containers are handled three at a time using slings and spreader beams when hoisting these from the railroad bay up the equipment hatch to the EL. 340' operating deck.

3/ Outer curtains weigh 1.68 tons each, inner weigh 1.64 tons.

TABLE A-5

IMPACT REGIONS
FOR AUXILIARY HOIST LOADS

REGION NO.	AREA
1	Reactor vessel and reactor cavity
2	Spent fuel pool
3	Reactor internals storage pit
4	Equipment hatch
5	Emergency condenser area
6	EL. 340' Floor

TABLE A-6

POTENTIAL SAFETY CONCERNS FOR AUXILIARY HOIST LOAD DROPS

NO.	SAFETY CONCERNS	APPLICABLE IMPACT REGIONS FOR NMP AUX. HOIST LOADS	RELATED NRC CRITERIA
1	Limit radiological dose at site bounary from impact of spent fuel to 1/4 of 10 CFR Part 100	Region 1 & 2	I
2	Limit K_{eff} to less than 0.95 due to crushing of fuel	Region 2	II
3	Maintain spent fuel pool cooling	Regions 2,3,4 & 6	III
4	Maintain extended core cooling - (reactor shutdown)	Regions 1,3,4 & 6	III, IV
5	Maintain capability to accomplish reactor shutdown and establish decay heat removal (plant operating)	Regions 3-6	IV

TABLE A-7

SUMMARY OF AUXILIARY HOIST LOAD DROP EVALUATIONS

REGION	APPLICABLE SAFETY CONCERNS	HEAVY LOADS	ANALYSES PERFORMED	ARE RESULTS IN COMPLIANCE WITH NUREG-0612	REMARKS	NRC RESPONSE ITEM
#1. Reactor cavity a. reactor head in place,	#4. Core Cooling	Reactor head insulation	Structural analysis of load drop onto head	Yes	Loads on the vessel skirt due to this drop are small. The energy of the load drop is significantly less than the energy absorbing capabilities of the reactor vessel. Additionally, although not considered in this evaluation, the head insulation structure would deform and absorb considerable energy for such a load drop.	2.2.4
b. reactor head removed, separator in place	#1. Dose	Portable radiation shield ("Japanese Bridge")	Structural analysis drop of shield onto separator assembly. This will bound effects of radiation shield drop or of a dryer/separator sling assembly drop	Yes	Separators would be deformed but would absorb energy; core supports would withstand loading; fuel would not be impacted.	2.2.4
		Vessel stud rack				
		Dryer/Separator				

REGION	APPLICABLE SAFETY CONCERNS	HEAVY LOADS	ANALYSES PERFORMED	ARE RESULTS IN COMPLIANCE WITH NUREG-0612	REMARKS	NRC RESPONSE ITEM
#1.Reactor Cavity (cont.) c.Drywell head in place; shield plugs removed	#4. Core Cooling	Small canal plug	Structural evaluation of loads on vessel skirt due to a drop of a small canal plug onto the drywell head	Yes	A small canal plug, if dropped, could tip and fall into the cavity and impact the drywell head. If the drywell head were penetrated, the vessel head could be impacted, resulting in loading of the vessel supports. Such loads on the vessel support skirt would be small.	2.24
#2.Spent Fuel Pool	#1. Dose #2. Criticality	Small canal Jib crane Fuel pool gates	A.Structural analysis of drop of each load to predict limiting number of fuel assemblies damaged B.Dose assessment C.Criticality evaluation	- No No	A large number of fuel pins could be damaged by a drop of the small canal plug into the pool, similarly a drop of the jib crane or fuel pool gates could lead to failure of several fuel assemblies. Resulting doses for the small canal plug drop do not satisfy NUREG-0612; dose levels for the jib crane drop or drop of the gates comply with NUREG 0612 evaluation criteria. The potential for criticality does not satisfy NUREG-0612. Inspection and identification procedures will be implemented to reduce the likelihood for a load drop.	2.2.4

REGION	APPLICABLE SAFETY CONCERNS	HEAVY LOADS	ANALYSES PERFORMED	ARE RESULTS IN COMPLIANCE WITH NUREG-0612	REMARKS	NRC RESPONSE ITEM
#2.Spent Fuel Pool (cont.)	#3.Fuel Pool cooling	Fuel pool gates	Structural evaluations	Yes	Other heavy loads such as the jib crane and the small canal plugs would impact fuel but would not directly impact the fuel pool liner. The gate drops would not result in gross liner penetration or failure; any leakage would be small.	2.2.4
#3.Reactor Internals storage pit	#3.Fuel pool cooling #4.Core cooling (plant shutdown) #5.Shutdown and core cooling (plant operating)	None identified (for auxiliary hoist)	Not applicable			2.3.2
#4.Equipment Hatch	#3.Fuel pool cooling; #4.Core cooling (reactor shutdown); #5.Shutdown and core cooling (plant operating)	New fuel in metal shipping containers (3 containers at once) New fuel channels in container	Structural analysis of load drop down equipment hatch of cluster of three fuel containers; fuel channels bounded by analysis of fuel container drop	Yes	The load drop analysis showed that no spalling, scabbing, perforation or structural failure would result for this load drop. The thick reinforced concrete floor in this area, which is provided to support the static load of a railcar and loaded shielded shipping casks, would be able to sustain the dropping of a cluster of three new fuel shipping containers loaded with fuel.	2.3.2

REGION	APPLICABLE SAFETY CONCERNS	HEAVY LOADS	ANALYSES PERFORMED	ARE RESULTS IN COMPLIANCE WITH NUREG-0612	REMARKS	NRC RESPONSE ITEM
#5.Platform over emergency condensers	#5.Shutdown and core decay heat removal (reactor operating)	Dryer/separator sling assembly Cask base plate	Structural analysis of drop of sling assembly and drop of cask base plate onto grating above emergency condensers	Yes	Load drop could cause some deformation of grating, but would not result in penetration or structural failure. Therefore, emergency condensers would not be impacted.	2.3.2
#6.El. 340' Floor	#3.Fuel pool cooling; #4.Core cooling (reactor shutdown) #5.Shutdown and core decay heat removal (reactor operating)	Portable radiation shield ("Japanese Bridge") Small canal plugs Fuel pool gate New fuel in containers Fuel channels in container Stud rack Jib crane Reactor head insulation Dryer/Separator sling assembly	Structural analysis of a small canal plug load drop onto the El. 340' floor. This bounds other load drops	No	A drop of a small canal plug from the 4 foot height required to clear the railings around the cavity would not result in perforation or structural failure, but could potentially result in scabbing of concrete from the bottom of the El. 340' floor. Similarly such scabbing of concrete could also result from a drop of the "Japanese Bridge" onto the El. 340' floor. Load drops of the other auxiliary hoist heavy loads in this region are not predicted to lead to scabbing. Inspection and identification procedures will be implemented to reduce the likelihood of a load drop.	2.3.2

Attachment B

STRUCTURAL EVALUATIONS OF POSTULATED LOAD DROPS

1.0 INTRODUCTION

Tables A-1 and A-4 of Attachment A list those heavy loads, that are carried by the handling systems evaluated in this report. Drop scenarios for loads handled by the auxiliary hoist include postulated drops onto the spent fuel pool floor, onto the spent fuel pool racks, onto various concrete slabs (e.g., refueling deck at elevation 340', and the equipment hatch floor), onto the steel grating platform over the emergency condensers, and into or onto the reactor vessel.

Assessments of the potential impact of a drop of the fuel transfer shield ("Japanese Bridge") onto the separator, and of the reactor head insulation onto the vessel head, were performed on the basis of a comparison of total available drop energy to elastic energy absorption capabilities. The structural evaluations for these postulated drops into the vessel cavity are presented in Section 2.0.

With the exception of the load drops into the spent fuel pool (see Section 6.0), the remaining structural load drop analyses involve determination of structural response of concrete floor slabs or steel grating to dynamic impact loadings. The spent fuel was evaluated to determine its capability to absorb the energy of the drop, without causing damage to the fuel. The heavy loads which could potentially be dropped onto various floor slabs were evaluated to identify loads which control local response (e.g., penetration, scabbing, spalling, perforation, etc.); loads that control overall structural response (e.g., large inelastic deformations or abrupt failures of principal structural members, etc.); and/or loads that may induce behavior that exhibits combined response such that either overall or local failure modes would control. The results of this evaluation are shown on Table B-1.

Where the controlling mode of response to postulated load drops is listed as "local", these loads were evaluated to determine the potential for slab penetration or perforation. Scabbing of the concrete deck backface was evaluated for all loads.

Postulated drops of the new fuel shipping container and small canal plugs fall in the category of loads which control local response, and bound other load drops that could potentially lead to local effects. A discussion of the local effects evaluation methodology can be found in Section 4.0.

Where the controlling mode of response is listed as "overall structural", these load drops were evaluated to determine the potential for producing gross and intolerable distortions of primary structural members and possibly propagating failures. Postulated drops of the spent fuel pool gate, the new fuel shipping containers, the fuel transfer shield, and various slot plugs fall in this category and bound other load drops that could potentially lead to "overall structural" effects. These analyses are discussed in Section 3.0.

TABLE B-1

SUMMARY OF CONTROLLING STRUCTURAL BEHAVIOR
RESULTING FROM POSTULATED HEAVY LOAD DROPS
(HEAVY LOADS HANDLED WITH THE REACTOR
BUILDING CRANE AUXILIARY HOIST)

Load	Approx. Weight (Tons)	Controlling Mode of Response: Overall Structural	Controlling Mode of Response: Local
1. Fuel Transfer Shield ("Japanese Bridge")	6.5	X	
2. Small Canal Slot Plugs	9.5/9.75	X	X
3. Fuel Pool Gates	0.5/0.65		X
4. Jib Crane	1.7		X
5. Reactor Head Insulation	5	X	
6. Vessel Stud Rack	1-2		X
7. New Fuel Containers	1		X
8. "Hydrolaser"	2	X	
9. Fuel Channel Containers	0.9		X
10. Dryer/Separator Sling Assembly	1.5		X
11. Auxiliary Hoist Load Block and Hook	1.7	N/A (See Section 5.0)	N/A
12. Cask Bottom Plate	1.5	X	X
13. LPRM Shipping Box	0.94		X
14. Curtains for Texas Tower	1.68/1.64		X

2.0 POSTULATED DROPS INTO THE VESSEL CAVITY

There are two potential consequences of interest which must be considered when evaluating load drops into the reactor vessel cavity. These are:

1) Loss of reactor vessel integrity; and

2) Evaluation for potential for fuel damage.

Evaluations for postulated drops into the vessel cavity were performed for the heavy loads handled by the auxiliary hoist which could potentially be carried over the vessel cavity. Those evaluations included drops of the reactor head insulation onto the vessel head and drops of the fuel transfer shield ("Japanese Bridge") onto the separator. The above drops were evaluated based on conservation of energy considerations. That is, the total energy of the postulated drops was compared to the energy absorbing capabilities of the supporting system.

The reactor head insulation weighs less than 5 tons. It was assumed to drop from its maximum carry height of 30 feet, onto the reactor vessel head. The resulting velocity of impact of this drop through air was calculated based on the equations of motion:

$$d = Vit + 1/2\ at^2$$

$$V = Vi + at$$

The energy on impact is calculated as:

$$E = 1/2\ MV^2$$

Therefore, the resultant energy of the postulated reactor head insulation drop onto the vessel head was determined to be 3.69×10^6 ft-lbs. The major portion of the impact load of the reactor head insulation is transmitted directly to the vessel flange. The load path is then through the Reactor Pressure Vessel (RPV) shell to the vessel supporting skirt which absorbs the impact. Based on an assessment of the vessel and its supports, the energy of the postulated drop is significantly less than the energy absorbing capabilities of the reactor vessel. Therefore, it was concluded that reactor vessel integrity would be maintained as a result of a postulated drop of the reactor head insulation onto the vessel head.

In order to assure that fuel integrity would be maintained for drops into the open vessel, an evaluation of a postulated drop of the fuel transfer shield onto the separator was performed. The fuel transfer shield weighs 6.5 tons, and was assumed to drop from its maximum carry height of 43 feet. In a manner similar to that described above for the reactor head insulation, the total energy of the fuel transfer shield drop was compared to the energy absorbing capabilities of the supporting system. Based on a comparison with the energy absorbing capabilities of the internals supporting system, it was concluded that no fuel damage would be predicted for a postulated drop of the fuel transfer shield onto the separator.

Based on the results of the analysis for the bounding drops into the cavity, reactor vessel integrity is predicted and no fuel damage is expected.

3.0 OVERALL STRUCTURAL RESPONSE EVALUATIONS

Notwithstanding the fact that the lifting systems, including the crane and slings, comply with the intent of applicable industry standards and possess demonstrated margins to failure, evaluations were performed for postulated drops of bounding heavy loads for a number of plant areas, including the spent fuel pool, the refueling deck at EL. 340' and the steel grating platform at EL. 359' over the Emergency Condensers.

A model of each location was developed with the objective of evaluating structural behavior for postulated flat and oblique drops of these loads from the normal carry heights.

A load drop methodology was developed to investigate the important modes of structural behavior. The objective of this methodology is to characterize structural behavior in terms of the available strain energy up to prescribed performance limits. These limits are dictated by either ductile or brittle modes of failure. The ductile mode is characterized by large inelastic deflections without complete collapse, while the brittle mode may result in partial failure or total collapse. The available internal strain energy that can be absorbed by the floor system without reaching those limits of unacceptable behavior is balanced against the externally applied energy resulting from a heavy load drop. It has been assumed that momentum is conserved and the kinetic energy of the drop drives the mass of the floor and induces strain. As an additional conservatism, no credit was taken for potential sources of energy dissipation such as concrete crushing and penetration.

For each impacted structural system, a model was developed and the response of the system to the dynamic impact loading was determined. The model was loaded in the direct vicinity of the drop location. This is considered to be conservative in view of the fact that the slabs or floors will help transfer load. The model was loaded until the moment capacity of any section or the allowable deflection was reached. For concrete, this moment capacity is defined by Chapter 10 of ACI 318-77 (Reference 8).

Generally, the ultimate load of a slab/grid system is reached prior to exceeding the hinge rotational capacity of particular sections provided that an unstable mechanism has not formed. This was found to be the case in this analysis. The hinge rotational capacity was used as a criterion to set the maximum allowable level of deflection for the slab/grid system. The hinge rotational capacity for concrete structures was developed in References 11 and 12 based on test results given in References 13 and 14 and is given as:

$$r_u = 0.0065\ (d/c) \leq 0.07 \qquad (1)$$

where,

r_u = rotational capacity of plastic hinge (radians)

d = distance from the compression face to the tensile reinforcement

c = distance from the compression face to the neutral axis at ultimate strength.

The maximum delfection for a beam with a plastic hinge at its center, is then given by:

$$Xm = (r_u L)/4 \qquad (2)$$

where,

Xm = maximum deflection

L = span of beam

Rotations of the magnitude governed by equation 1 result in cracking which is confined to a region below (above) the tensile reinforcement. Generally speaking the section will remain intact with no crushing, spalling or scabbing due to flexure. However, scabbing may occurr as a result of shock wave motion associated with the reflection of tensile waves from the rear surface or shear plug formation. The potential for scabbing was evaluated for all load drops, and is discussed in Section 4.0, following.

The load/deflection history up to the point of the ultimate loading, coupled with the maximum allowable deflection, defines the maximum level of strain energy absorption, provided that a shear failure has not occurred. The shear stress at limiting sections was checked and compared to allowables as specified in Chapter 11 of ACI 318-77 (Reference 8).

For each area where the potential for overall structural response modes was considered possible, an assessment of the bounding drop was made. The criterion for selection was impact energy of the postulated drop.

For example, for heavy loads carried over the refueling deck at elevation 340', a drop of the 9.75 ton small canal slot plugs onto the floor bounds other postulated heavy load drops. Therefore, the overall response of a postulated canal slot plug drop, from its normal carry height onto the floor, was analyzed. The results of the analysis indicate that the impact energy of the drop is less than the energy absorbing capacity of the refueling deck floor system. Therefore, no overall structural failure is predicted.

Similarly, the potential impact to the platform over the Emergency Condensers was evaluated and found to be bounded by a postulated drop of the cask base plate from its normal carry height. The ultimate load capability of the grating was determined assuming a one directional bending. This is considered conservative since the 1.5 ton cask base plate has a radius of 55". The results of the analysis indicate that the steel grating and supporting beams are capable of withstanding the impact of the postulated drop, with no overall structural failure expected.

In addition to the conservatisms previously mentioned, the following conservatisms are also inherent in the methodology used in the evaluation:

1) Static material strengths for concrete and steel were generally used. Test data show that this property increases with the increased strain rates associated with dynamic loadings. For example, References 15 and 16 recommend dynamic increase factors of 1.25 for the compressive strength of concrete and 1.20 for the flexural, tensile and compressive strength of structural steel.

2) Design (minimum) material properties for concrete and steel were used. No increase was taken for the aging of concrete which can amount to a factor of up to 1.35 (Reference 17) of increased strength. Also, the average strength for structural steel is nearly a factor of 1.25 (Reference 18) higher than the minimum yield requirement specified by ASTM. While these factors above minimum code strength exist and contribute to structural margins, they were not used in the evaluation.

3) Equation 1 for hinge rotational capacity was used. This corresponds to support rotations of the order of 2 degrees with minimum cracking and no crushing or scabbing. To meet necessary performance requirements (i.e. halting propagating failures), larger rotations in the range of 5 to 12 degrees could be tolerated. Such rotations would lead to crushing, spalling and scabbing of the section (Reference 16); however, overall load carrying capability is expected to remain intact. Experimental observations (Reference 19) suggest even further capability for well designed and well anchored slabs. Failure modes at such levels initially appear to be controlled by yielding in shear and flexure followed by membrane stretching until failure occurs, normally at the support edge of the slab. Use of these larger rotational capabilities would have resulted in greater energy absorbing capabilities of the grid system.

4) The analysis used ACI 318-77 allowable shear stresses. A significant body of data suggests the existence of higher shear capabilities on the order of $10\sqrt{f'c}$ to $20\sqrt{f'c}$ (References 20-28). It is expected that the shear capabilities for these beams would tend to be in the higher end of the range since the majority of the beams are "deep". Deep beams behave as tied arches with significant reserve capacity.

5) In many cases, the analysis neglected the two-way resistance capability of the slab. It is expected that the slab would contribute increased strength particularly at larger deformations.

6) The load was distributed directly under the dropped shield plug. In reality a more favorable load distribution would exist due to the load distribution capability of the slab.

7) No credit was taken for local energy dissipation associated with any crushing of the shield plug or the immediate surface of the floor.

Based upon the results of the above analyses for overall structural response, the carry heights associated with the normal operating procedures were found to be acceptable.

4.0 LOCAL STRUCTURAL RESPONSE EVALUATIONS

Selected loads such as the small canal slot plugs and various equipment which are carried over concrete slabs were evaluated to assess the acceptability and potential consequences of postulated drops from normal carry heights. The acceptance criteria were based on the capability of the concrete slabs to resist perforation, penetration, and underside scabbing.

Procedures recommended in References 9 and 10 were followed. The modified National Defense Research Committee (NDRC) formula (Reference 35) was chosen because it has been shown to give the best fit with available experimental data (References 1 and 36). The NDRC formula for the depth of penetration, x (inches), of a solid cylindrical missile is given by:

$$x = (4\ KNWd(V/1000d)^{1.8})^{1/2} \text{ for } x/d \leq 2.0 \qquad (3)$$

or

$$x = (KNW(V/1000d)^{1.8}+d \text{ for } x/d > 2.0 \qquad (4)$$

where W = weight of the missile (pounds)
d = diameter of missile (inches)
V = impact velocity of missile (feet/second)
N = missile shape factor
= 0.72 flat-nosed missiles
= 0.84 blunt-nosed missiles
= 1.00 spherical-nosed missiles
= 1.14 sharp-nosed missiles
K = concrete penetrability factor
= 180/   f'c (f'c = concrete compressive strength pounds/square inch)

The thickness of reinforced concrete needed to resist impact without perforation and scabbing are given by the following Army Corps of Engineers formulae which can be used in conjunction with equations 3 and 4 (Reference 2).

$$ts/d = 2.12 + 1.36\ (x/d) \quad \text{for} \quad 0.65 \leq x/d \leq 11.75 \qquad (5)$$

$$tp/d = 1.32 + 1.24\ (x/d) \quad \text{for} \quad 1.35 \leq x/d \leq 13.5 \qquad (6)$$

where

ts = concrete thickness required to prevent scabbing

tp = concrete thickness required to prevent perforation

Equations 5 and 6 were later extrapolated for small values of x/d (Reference 39) giving,

$$ts/d = 7.91\ (x/d) - 5.06\ (x/d)^2 \quad \text{for} \quad x/d \leq 0.65 \qquad (7)$$

$$tp/d = 3.19\ (x/d) - 0.718\ (x/d)^2 \quad \text{for} \quad x/d \leq 1.35 \qquad (8)$$

A 10 percent margin on thickness has been applied in the use of equations 5 thru 8 as recommended in Reference 9.

Limited penetration and scabbing was predicted for the set of bounding heavy load drops considered. However, in no case were the concrete slabs predicted to be perforated for normal carry heights.

5.0 REACTOR BUILDING CRANE AUXILIARY HOIST LOAD BLOCK AND HOOK

NUREG-0612 requires that the load block and hook be considered as a heavy load. For the main hoist the load block and hook are carried by redundant load paths back to the trolley frame. Accordingly a load drop of the main hoist load block and hook is not considered feasible. For the auxiliary hoist, the load block is used for handling numerous loads, including the

reactor vessel head insulation, small canal plugs, and the jib crane. In moving these loads, the auxiliary hoist hook, load block, rope, drum, sheave assembly, motor shafts, gears, and other load bearing members are subjected to increased stresses. By comparison, these components are subjected to a considerably smaller load when only the hook and load block are being moved. Based on this, it is not considered feasible to postulate a random mechanical failure of the auxiliary hoist load bearing components when moving the auxiliary hoist load block without a load. As stated earlier, the auxiliary hoist is provided with redundant limit switches and a 150% full motor torque brake. These features reduce the potential for a two blocking event or an uncontrolled lowering of the load. Therefore, a drop of the auxiliary hoist load block and hook is considered extremely remote and no analyses was performed.

6.0 STRUCTURAL ANALYSIS OF LOAD DROPS ONTO FUEL IN THE SPENT FUEL POOL

The evaluation of heavy load handling operations over the spent fuel pool included an analysis of postulated drops onto the spent fuel racks. Based on a consideration of weight, size, and impact energy, a postulated drop of any of the three small canal plugs was determined to control (with respect to potential for damage and offsite doses). The analysis was performed to determine the capability of the fuel bundles to absorb the energy of the postulated small canal plug drop, without causing damage to the fuel. Although the plug drops onto the racks, the limiting consideration is the ability of the fuel rods within the fuel channel to absorb energy while maintaining fuel integrity.

The resistance of a fuel bundle was determined based on the buckling strength and deflection of individual fuel rods. For conservatism, it was assumed that all rods within a fuel bundle buckle in the same direction. Based on a maximum lateral deflection of 2.546", the fuel rod stress and strain for the buckling load of 408 lbs. were determined to exceed allowables. Therefore, the response of the fuel rods was reevaulated to determine the magnitude of deflection, stress, and strain at the point where the fuel rod response has reached allowable levels.

Based on an acceptance criterion of 1% strain, an individual fuel rod was determined to be capable of absorbing approximately 50 ft.-lbs. of energy. An assessment of the impact energy for the postulated small canal plug drop was then performed. It was found that, based on conservative assumptions, the small canal plug could impact 140 fuel bundles at 1% strain was determined to be 434,000 ft.-lbs., the possibility of damage to fuel could not be precluded for a straight drop of the gate onto 140 fuel bundles. These values were calculated assuming the current spent fuel racks with the highest density were the ones impacted, and that they were filled with 8x8 assemblies (62 fuel rods, 2 water rods each).

No additional evaluation of plug and fuel dynamics and behavior was attempted. However, it is estimated that damage, in addition to that suffered by the 140 assemblies, could occur as impacted fuel assemblies buckle and cause damage to contiguous assemblies. Also, because the 140 assembly damage calculation was shown to be limiting among the cases studied, no additional calculation was performed of damage from the straight drop of the plug (in a vertical position), followed by rotation and topple-over onto additional assemblies.

Analyses of jib crane drops or drops of transfer gates using the same methodology showed that considerably fewer fuel assemblies would be impacted.

In summary, an evaluation was performed of postulated drops of all potential loads onto the fuel in the spent fuel pool. The controlling load, from a standpoint of fuel damage, was any of the 3 small canal plugs. The calculations did not model additional damage that could occur from fuel assembly buckling and subsequent damage to adjacent assemblies.

References

1. Beth, R.A. and Stipe, J.G., "Penetration and Explosion Tests on Concrete Slabs", CPPAB Interim Report No. 20, January 1943.

2. Beth, R.A., "Concrete Penetration" OSRD-4856, National Defense Research Committee Report A-319, March 1945.

3. Letter from Niagara Mohawk to NRC dated July 28, 1981 (first report concerning heavy loads evaluation).

4. ANSI B30.9-1971, Slings, American National Standard, Safety Standards for Derricks, Hoists, Hooks, Jacks and Slings, American Society of Mechanical Engineers, 1971.

5. CMAA 70-1975, Specifications for Electric Overhead Traveling Cranes, Crane Manufacturers Association of America, 1975.

6. ANSI 30.2-1976, Overhead and Gantry Cranes, American National Standards, Safety Standards for Cableways, Cranes, Derricks, Hoists, Hooks, Jacks and Slings, American Society of Mechanical Engineers, 1976.

7. "ICES STRUDL, Structural Design Language", McDonnell Douglas Automation Company, April 1981.

8. Building Code Requirements for Reinforced Concrete, ACI 318-77, American Concrete Institute, December 1977.

9. Civil Engineering and Nuclear Power, Report of the ASCE Committee on Impactive and Impulsive Loads, Vol. V, American Society of Civil Engineers, September 1980.

10. Structural Analysis and Design of Nuclear Plant Facilities, American Society of Civil Engineers, 1980.

11. ACI 349-76, Code Requirements for Nuclear Safety-Related Concrete Structures, Appendix C - "Special Provisions for Impluse and Impactive Effects", American Concrete Institute, 1976.

12. Kennedy, R.P., "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects", Journal of Nuclear Engineering and Design, Vol. 37, No. 2, May 1976.

13. Mattock, A.H., "Rotational Capacity of Hinging Region in Reinforced Concrete Beams", Flexural Mechanics of Reinforced Concrete, ASCE 1965-50 (ACI SP-12), American Society of Civil Engineers, 1965.

14. Corely, W.G., "Rotational Capacity of Reinforced Concrete Beams", Journal of Structural Division, ASCE, Vol. 92, No. ST5, Proc. Paper 4939, Oct. 1976, pp. 121-146.

15. "Design of Structures for Missile Impact", Topical Report BC-TOP-9A, Bechtel Power Corporation, September 1974.

16. Structures to Resist the Effects of Accidental Explosions, TM5-1300, Department of the Army, Washington, D.C., July 1965.

17. Neville, A.M., Properties of Concrete, J. Wiley & Sons, New York, 1975.

18. Design of Structures to Resist the Effects of Atomic Weapons - Strength of Materials and Structural Elements, TM5-856-2, Department of the Army, Washington, D.C., August 1965.

19. Personal communication between Professor William J. Hall and Howard A. Levin, October 5, 1981.

20. Wang, C.K. and Salmon, C.G., Reinforced Concrete Design, Intext Educational Publishers, New York, 1973.

21. Ferguson, P.M., Reinforced Concrete Fundamentals, J. Wiley, New York, 1973.

22. Untrauer, R.E. and C.P. Seiss, "Strength and Behavior in Flexure of Deep Reinforced Concrete Beams Under Static and Dynamic Loading", Civil Engineering Studies Structural Research Series Report No. 230, University of Illinois, Urbana, October 1961.

23. Austin, W.J., et al, "An Investigation of the Behavior of Deep Members of Reinforced Concrete and Steel", Civil Engineering Studies Structural Research Series No. 187, University of Illinois, Urbana, January 1960.

24. de Pavia, H.A.R. and C.P. Siess, "Strength and Behavior in Shear of Deep Reinforced Concrete Beams Under Static and Dynamic Loading", Civil Engineering Studies Structural Research Series Report No. 231, University of Illinois, Urbana, Oct. 1961.

25. de Pavia, H.A.R., and W.J. Austin, "Behavior and Design of Deep Structural Members -- Part 3 -- Tests of Reinforced Concrete Deep Beams, "Civil Engineering Studies Structural Research Series No. 1974, University of Illinois, Urbana, March 1960.

26. Winemiller, J.R. and W.J. Austin, "Behavior and Design of Deep Structural Members -- Part 2 -- Tests of Reinforced Concrete Deep Members with Web and Compression Reinforcement", Civil Engineering Studies Structural Research Series Report No. 193, University of Illinois, Urbana, August 1960.

27. Newmark, N.M. and J.D. Haltiwanger, "Air Force Design Manual -- Principles and Practices for Design of Hardened Structures", AFSWC-TDR-62-138, December 1962.

28. Crawford, R.E., et al, "The Air Force Manual for Design and Analysis of Hardened Structures", AFWL-TR-74-102, October 1974.

29. Timoshenko, S.P., and Goodier, J.N., Theory of Elasticity, McGraw-Hill Book Company, New York, 1970.

30. Baker, Kovalevsky, and Risk, Structural Analysis of Shells, McGraw Hill, New York, 1972.

31. Reissner, "On the Theory of Thin Elastic Shells", 1949.

32. Roark and Young, Formulations for Stress and Strain, McGraw-Hill, New York, 1975.

33. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, USNRC, July 1980.

34. Effects of Impact and Explosion, Summary Technical Report of Division of 2, National Defense Research Committee, Vol. 1, Washington, D.C., 1946.

35. Vassallo, F.A., Missile Impact Testing of Reinforced Concrete Panels, HC-5609-D-1, Calspan Corporation, January 1975.

36. Stephenson, A.E., "Full Scale Tornado Missile Impact Tests", Electric Power Research Institute, Final Report NP-440, July 1977.