SNUPPS

Standardized Nuclear Unit
Power Plant System

5 Choke Cherry Road
Rockville, Maryland 20850
(301) 869-8010

Nicholas A. Petrick
Executive Director

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SUBJ: Control of Heavy Loads

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Docket No. STN 50-482 and STN 50-483

References: 1. SLNRC 82-0033, dated August 4, 1982: Same Subject
2. SLNRC 84-0008, dated January 27, 1984: Same Subject

Dear Mr. Denton:

Reference 1 provided 15 copies of the SNUPPS REPORT ON CONTROL OF HEAVY LOADS. Revision 1 to the report was provided in Reference 2. Enclosed are 15 sets of revised pages which constitute Revision 2 to the report. Revision 2 incorporates minor changes to the body of the report, a revised Appendix C, "Analyses of Reactor Vessel Head Drop", which was updated by Revision 14 to the SNUPPS FSAR, and provides a new Appendix E, "Evaluation of SNUPPS Reactor Vessel Head and Internals Lifting Devices" which is a summary of a Westinghouse analysis of special lifting devices.

Very truly yours,

NA Petrick

Nicholas A. Petrick

MHF/nld9a26
Enclosure

cc: G. L. Koester KGE
D. T. McPhee KCPL
D. F. Schnell UE
J. Neisler/B. Little USNRC/CAL
W. Schum/A. Smith USNRC/WC
J. Konklin USNRC/RIII

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PDR ADOCK 05000482
A PDR

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Miscellaneous Hoists

Other load handling operations which are not part of the fuel handling system include miscellaneous hoists and monorails in the reactor, auxiliary, fuel, control, and diesel generator buildings. The hoists' travel paths are restricted to the monorail layout by mechanical stops.

Table 2 identifies primary loads lifted and hoist capacities for the miscellaneous hoists. No special lifting devices are used with the miscellaneous hoists. Lifting devices for these hoists are in accordance with guidance provided by ANSI B30.9-1971, as clarified by NUREG-0612, Section 5.1.1.(5).

Tables 1 and 2 provide a listing of each applicable crane or hoist and loads to be handled. Lifting devices are described below.

Certain of the typical PWR heavy loads identified in NUREG-0612 Table 3.1-1 are not applicable to SNUPPS and are not covered by Tables 1 and 2. The SNUPPS missile shield is mounted on a dedicated transfer cart that shares the refueling machine rails and is not normally handled by a crane. In addition, certain heavy loads such as the spent fuel cask have not been specifically designed for SNUPPS at this time; however, load values that envelope the expected weight of these items are included in the evaluation. Compliance with ANSI N14.6-1978 for the cask lifting device will be provided.

The supplier (Westinghouse) has performed analyses of the reactor vessel head lifting device (HKE02), and reactor vessel internals lifting device (HKE03), with regard to compliance with ANSI N14.6-1978. The results of these analyses are summarized in Appendix E.

Written procedures that reflect the results of this evaluation will be developed to govern the handling of all heavy loads whose drop could damage fuel or safe shutdown equipment. These procedures will incorporate the requirements provided in Section 5.1.1.(2) of NUREG-0612.

Plant procedures will be developed for inspection, testing, and maintenance of those cranes identified in Table 1. These procedures will include the guidance provided by Chapter 2-2 of ANSI B30.2-1976, as clarified in NUREG-0612, Paragraph 5.1.1.(6) with regard to frequency of inspections, tests, and maintenance.

Regarding inspection frequencies, the SNUPPS cranes are classified as standby cranes which are not exposed to adverse conditions. Therefore, inspections will be done in accordance with Section 2-2.1.4.b and Section 2-2.1.3, Periodic Inspections, as modified by Paragraph 5.1.1(6) of NUREG-0612. The frequent visual inspection guidelines of Section 2-2.1.2 will be followed during periods of use except for item 2 concerning limit switches. The main upper limit switch will be subject to the requirements of Section 2-3.2.4 at the beginning of each shift when the cranes are in use.

Miscellaneous hoists identified in Table 2 will be inspected, tested, and maintained based on the manufacturer's recommendations.

Specific plant procedures will be developed that address crane operator training, qualification, and conduct for those cranes identified in Table 1. These procedures will incorporate the guidance provided by ANSI B30.2-1976, Chapter 2.3.

The containment polar crane, the cask handling crane, the spent fuel pool bridge crane, and the refueling machine are designed to the standards of CMAA No. 70 (1975). The SNUPPS cranes were ordered in 1974, and their purchase specifications included reference to ANSI B30.2-1967 for design requirements.

The miscellaneous hoists throughout the plant are designed in accordance with the requirements of ANSI B30.10-1975, Hooks; ANSI B30.11-1973, Monorail Systems and Underhung Cranes; and ANSI B30.16-1973, Safety Standards for Overhead Hoists.

[Paragraphs on pp. 12-13 were rearranged to more logical sequence]

APPENDIX C

ANALYSES OF REACTOR VESSEL HEAD DROP

In a head assembly removal or reassembly, it is postulated that the polar crane fails. If this unlikely event should occur, various consequences would prevail depending upon the position of the vessel head assembly in relation to the reactor vessel at the time of the polar crane cable failure. Six accident cases have been defined that envelope the effects of dropping the vessel head assembly at critical points along the path from the reactor vessel to the head storage stand.

The accident cases that have been defined are as follows:

Case I - Head assembly falls approximately 14 feet through air while engaged on guide studs and impacts the vessel flange.

Case II - Head assembly falls 4 feet through air, 24 feet through water, and impacts the vessel flange.

Case III- Head assembly falls 4 feet through air, 6.5 feet through water, strikes the guide studs, falls 18 feet through water and impacts the vessel flange at an angle.

Case IV - Head assembly falls 4 feet through air, 24 feet through water, and lands partially on the reactor vessel flange and partially on the concrete.

Case V - Head assembly falls 4 feet through air, 24.5 feet through water, and strikes the floor of the refueling pool.

Case VI - Head assembly hits refueling pool wall, rotates into the refueling pool, and falls 24.5 feet through water; vessel head closure flange strikes the concrete.

The above six cases were evaluated generically in Reference 1. It was found that Case II is the limiting accident case in terms of maximum impact velocity. In Part I of the SNUPPS head drop accident analysis, the reactor vessel nozzles are analyzed for a Case II situation using the methods and assumptions of Reference 1. The necessary changes from Reference 1 in weights, stiffnesses, and drop heights were made to make this analysis more specific to SNUPPS while remaining conservative in terms of reactor vessel nozzle evaluation.

In order to more fully address the question of maintenance of core cooling capability, a reactor vessel support evaluation for the postulated head drop accident was also performed and is reported as Part II of the SNUPPS head drop accident analysis.

Part I - Reactor Vessel Nozzle Evaluation

Case II - Head assembly falls 4 feet through air, 24 feet through water, and impacts the vessel flange.

During reactor disassembly or reassembly, the vessel head is positioned 28 feet above the vessel flange, while the water depth is 24 feet. When the head assembly is directly above the reactor vessel and at the maximum lift height, the polar crane cable is postulated to fail. The head assembly falls, engages on the guide studs, and lands directly on the reactor vessel flange.

a. Head assembly impact velocity calculation

Assumptions

1. Final velocity is assumed to be equivalent to that of a 28-foot drop through water.

2. Only half of the buoyant force is taken into account since none of the head assembly is in the water at the beginning of the drop.

3. Drag coefficient is that of a flat-surfaced hemisphere.

4. The head does not bind with the guide studs.

Analysis

The general equation for acceleration through a liquid is (Reference 2):

$$(W-B) - C_D \rho_w (A/2)(dy/dt)^2 = md^2y/dt^2$$

where:

W = weight of integrated reactor vessel head

B = weight of displaced water

ρ_w = density of water

C_D = drag coefficient

A = projected area of object

m = mass of object

v = velocity of object

Integrating results in the equation for velocity:

$$v = \sqrt{\frac{C_1}{C_2}} \cdot \frac{e^{t/C_3} - 1}{e^{t/C_3} + 1}$$

$C_1 = W-B$

$C_2 = C_D \rho_w A/2$

$C_3 = m/(2\sqrt{C_1 C_2})$

Integrating results in the equation for distance:

$$y = \sqrt{\frac{C_1}{C_2}}\ [2\ C_3\ \ln\ (\frac{e^{t/C_3} + 1}{2}) - t]$$

The parameter values for this problem are:

$W = 359{,}329$ lbs (Table C-1)

$B = \frac{\rho_w}{2\rho s} W = 22{,}918$ lbs

$\rho_w = 0.0361$ lbs/in.^{3} = 1.937 slugs/ft^3

$\rho_{steel} = 0.283$ lbs/in.^{3}

$C_D = 1.17$ (Reference 3)

$A = 229$ ft^2

$m = 11{,}159$ slugs

$C_1 = 336{,}411$ lbs

$C_2 = 259.5$ lbs·sec^2/ft^2

$C_3 = 0.597$ second

For y = 28 feet, collision occurs after 1.515 seconds.

At that time, the velocity is 368 inches/second.

b. Consideration of fuel assemblies

The fuel assemblies, and specifically the fuel cladding, must retain their integrity in order to ensure no release of fission-product gases. During this accident, the head assembly itself does not come in contact with the fuel assemblies.

The drive rods, which extend above the reactor vessel flange, are carefully inserted into the head during normal refueling operations. However, during the accident, it cannot be assumed that all the drive rods enter the head penetrations. The drive rods will buckle under the weight of the falling head, and the buckling load of each buckling drive rod must be able to be withstood by the corresponding fuel assembly. The drive rod buckling load is the only major force experienced by the fuel assemblies and is calculated hereafter.

Model of Critical Buckling Load for the Drive Rod

Refer to Figure C-1

1. Buckling Load of Section 1

   Before the buckling load of Section 1 can be calculated, the end condition at its base must be defined. The end condition for Section 1 will be determined by the reaction and buckling load of Section 2. Section 2 will be considered to have 2 pinned ends because of the small radial clearance (0.325 inch).

2. Buckling Load for Section 2 (refer to Figure C-2)

   $$P_{cr} = \frac{\pi^2 EI}{\ell^2}$$

   P_{cr} = critical buckling load

   E = modulus of elasticity

   I = moment of inertia of an area

   ℓ = length

   D = average of major and minor drive rod thread diameters

   d = inside diameter of drive rod

   Calculate I

   $$I = \frac{\pi (D^4 - d^4)}{64}$$

   $$D = \frac{1.75 + 1.475}{2} = 1.613$$

   $$I = \frac{\pi(1.613^4 - 0.875^4)}{64} = 0.3031 \text{ in.}^4$$

   $$E = 28.3 \times 10^6 \text{ lb/in.}^2$$

   $$\ell = 12.50$$

   $$P_{cr} = \frac{\pi^2 (28.3 \times 10^6)(0.3031)}{(12.50)^2}$$

   $$P_{cr} = 541{,}816 \text{ lbs}$$

   The buckling load for Section 2 is 541,816 pounds. A P_{cr} load for Section 1 of anythig less than 541,816

pounds will indicate that the two act independently of each other and Section 2 will not buckle, therefore Section 1 will be considered to have a clamped end and a free end (refer to Figure C-3)

$$P_{cr} = \frac{\pi^2 EI}{4\ell^2}$$

$$E = 28.3 \times 10^6 \text{ lb/in.}^2$$

$$I = 0.3031 \text{ in.}^4$$

$$P_{cr} = \frac{\pi^2 \ (28.3 \times 10^6) \ (0.3031)}{4(148.75)^2}$$

$$P_{cr} = 956.5 \text{ lbs}$$

The maximum vertical force on the fuel assembly is the buckling load of Section 1. An impact force of this value will impart no damage to the fuel assembly, and fuel cladding integrity will be maintained.

c. Consideration of reactor vessel nozzles

Description

The impact load of the head assembly on the vessel is transmitted through the vessel to the four supported vessel nozzles. The nozzles must be able to support this load without exceeding the allowable stress limits. The effects on the nozzles were evaluated by conservatively assuming the head drop through 28 feet of air.

Assumptions

1. The head assembly is assumed to drop 28 feet through air.

2. If it is assumed that the stresses due to the impact load are distributed throughout any elastic body exactly as in the case of static loading, then it can be shown that the vertical deformation δ_i and the stresses σ_i produced in any such body by the vertical impact of a body falling from a height (h) are greater than the deformation δ and stress σ produced by the

weight of the same body applied as a static load in the ratio (Reference 4):

$$\frac{\delta_i}{\delta} = \frac{\sigma_i}{\sigma} = 1 + \sqrt{1 + 2\frac{h}{\delta}}$$

If h=0, we have the case of sudden loading and $\frac{\delta_i}{\delta} = \frac{\sigma_i}{\sigma} = 2$ as assumed.

The above approximate formula is derived on the assumption that the impact load strains the elastic body in the same way (though not in the same degree) as static loading and that all the kinetic energy of the moving body is expended in producing this strain.

Actually in the impact some kinetic energy is dissipated and this loss, which can be found by equating the momentum of the entire system before and after impact, is more conveniently taken into account by multiplying the available energy by a factor K, the value of which is as follows (Reference 4):

Energy Dissipation Factor
$$K = \frac{1 + \frac{1}{3}\frac{M_1}{M}}{\left(1 + \frac{1}{2}\frac{M_1}{M}\right)^2}$$

where:

M = mass of the moving body = $\frac{W}{g}$

M_1 = mass of the body struck by the moving body $= \frac{W_1}{g}$

From the above equations, the impact load W_i can be derived as follows:

$$W_i = W\left(1 + \sqrt{1 + \frac{2Kh}{\delta}}\right)$$

3. The rigidity of the vessel flange causes the impact loads to be distributed evenly to the four supporting nozzles.

4. The reactor vessel is supported by two inlet and two outlet nozzles.

5. Load deformation of the head at impact is neglected.

6. The area and moment of inertia for the inlet nozzle are larger than for the outlet nozzle at the nozzle to shell juncture region. Similar difference exists also for the cross section at the integral pad location. Hence, the outlet nozzle was evaluated for impact stresses.

<u>Analysis</u>

Determination of the Impact Load W_i

The head upper package and reactor vessel can be idealized as a simple spring mass system as shown in Figure C-4.

Determination of Spring Constant k_v

The upper portion of the reactor vessel was idealized as spring k_v. To simplify the analysis, the upper portion of the vessel was conservatively assumed to be a cylindrical member with the cross section and parameters as follows (see Figure C-5):

$$\delta = \frac{PL}{AE}$$

$$k_v = \frac{P}{\varepsilon} = \frac{AE}{L}$$

where:

R = outside radius

r = inside radius

t = thickness

A = area

L = length

E = modulus of elasticity for carbon molysteel at 70 F

$$A = \pi(R^2 - r^2)$$

$$= \pi\ (96.19^2 - 85.4^2) = 6134 \text{ in.}^2$$

$$k_v = \frac{(6134)\ (29.9 \times 10^6)}{106.5} = 1.722 \times 10^9 \text{ lb/in.}$$

Spring constants for the inlet (k_{in}) and outlet (k_{on}) regions were determined from a 3-D finite element analysis of the reactor vessel.

Determination of equivalent spring constant k_e of the system shown in Figure C-4:

$k_{in} = 79.8 \times 10^6$ lbs/in.

$k_{on} = 71.7 \times 10^6$ lbs/in.

$k_s = 25.3 \times 10^6$ lbs/in.

$k_v = 1.722 \times 10^9$ lbs/in.

For the nozzles and supports in series:

$$k_{inlet\text{-}support} = \frac{1}{1/k_{in} + 1/k_s} = k_{ins}$$

$$k_{inlet\text{-}support} = 1.92 \times 10^7 \text{ lbs/in.} = k_{ins}$$

$$k_{outlet\text{-}support} = \frac{1}{\frac{1}{k_{on}} + \frac{1}{k_s}}$$

$$k_{outlet\text{-}support} = 1.87 \times 10^7 \text{ lbs/in.} = k_{ons}$$

For springs in parallel:

$$k_p = 2\ k_{ons} + 2\ k_{ins}$$

$$k_p = 7.58 \times 10^7 \text{ lbs/in.}$$

For the springs in series:

$$k_e = \frac{1}{1/k_v + 1/k_p} = 7.26 \times 10^7$$

The weight of the upper package, head assembly, and crane block (W) is 359,329 pounds.

The weight of the vessel flange and nozzle shell (W_1) is 290,000 pounds.

$$K = \frac{1 + 1/3 \; \frac{M_1}{M}}{\left(1 + 1/2 \; \frac{M_1}{M}\right)^2} = \frac{1 + 1/3 \; \frac{290,000}{359,329}}{\left[1 + 1/2 \left(\frac{290,000}{359,329}\right)\right]^2} = 0.644$$

Energy dissipation factor (K) is 0.644.

$$\delta = \frac{W}{k_e} = 0.00495 \text{ in.}$$

Static deflection (δ) is 0.00495 in.

Impact Load

Equation $W_i = W\left(1 + \sqrt{1 + \frac{2Kh}{\delta}}\right)$ becomes:

$$W_i = 359,329 \left(1 + \sqrt{1 + \frac{(2)(0.644)(28)(12)}{0.00495}}\right)$$

Impact load = 106.6×10^6 lbs

Assuming a perfect drop and four supported nozzles equally share the impact load:

Impact force/nozzle = 26.7×10^6 lbs

Determine the stress developed in the outlet nozzle due to impact load (refer to Figure C-6).

R = 22.5 in.

r = 14.5 in.

Moment (I) = $\frac{\pi}{4}(R^4 - r^4) = 1.67 \times 10^5$

Maximum Bending Stress

$$\sigma_B = \frac{MR}{I} = 51,477 \text{ psi}$$

where:

M = (impact force/nozzle) $\cdot$ (a) = $(26.7 \times 10^6) \cdot (14.31)$

Shear Stress

Maximum load at the nozzle cross section = impact force/nozzle = 26.7 x 106 lbs.

$$\tau_{avg} = \frac{\text{Impact Force/Nozzle}}{\pi\ (R^2 - r^2)} = \frac{26.7 \times 10^6}{\pi (22.5^2 - 14.5^2)}$$

$$= 28{,}712 \text{ psi}$$

Maximum Principal Stress

$$\sigma_{max} = \frac{1}{2}\,\sigma_B + \sqrt{\frac{1}{4}\,\sigma_B^2 + \tau_{avg}^2} = 64{,}298 \text{ psi}$$

Therefore, σ_{max} is less than the allowable limit of 84,000 psi (lesser of $3.6S_m$ or $1.05S_u$).

The velocity of the head assembly falling 28 feet through air is 510 in./sec. This, compared with the more realistic velocity of 368 in./sec calculated, reflects one of several conservatisms used in the evaluation of the nozzles.

The results of the preceding analysis show that the reactor vessel nozzles are not stressed above allowable limits. In order to more completely address the question of maintenance of core cooling capability, an evaluation of the reactor vessel supports was performed, and the results are reported in Part II.

d. Consideration of core barrel

In a normal reassembly of the reactor vessel, the head assembly first contacts the upper internals flange applying pressure against the core barrel holddown spring. The upper internals depress the holddown spring until the head assembly contacts the vessel flange. During this accident case the above reassembly description occurs compressing the hold-down spring. Any amplified effects could cause some yielding on the outer portion of the core barrel and upper internals flanges.

The bottom of the core barrel is designed with supports for a hypothetical accident in which the core barrel support, the flange, might fail and allow the core barrel to fall. These supports will limit its travel to approximately 1-1/4 inches in a cold condition without any failure to the fuel. Therefore, in an unlikely event of this accident case causing failure of the core barrel the lower internals supports would

limit the core barrel travel as in the hypothetical accident to approximately 1-1/4 inches and still maintain the integrity of the core.

Part II - Reactor Vessel Support Evaluation

Due to the high impact loads on the reactor vessel nozzles reported in Part I, a reactor vessel support evaluation was performed for the effects caused by a postulated vessel head drop accident In an effort to reduce the probability of vessel support damage due to high impact loads, the accident evaluation input parameters were revised to more closely reflect actual SNUPPS plant conditions during refueling. These changes are described as follows.

a. The drop distance of the reactor vessel head package of Part I was assumed to be 28 feet entirely through air. The SNUPPS plant layout, however, allows the head to be moved laterally when the vessel head guide studs, which are about 18 feet long, have been cleared. Therefore, during removal, when the vessel head is followed with water, the maximum drop through water is about 18 feet. During replacement, the head is again followed with water while being lowered onto the guide studs down to a height of about 14 feet. At this point the head is held steady and the refueling canal is drained. The maximum drop through air is therefore 14 feet. This is also the most limiting case in terms of impact velocity.

b. The mass assumed to resist the impact of the falling head assembly was given in Part I as 290 kips. This is the weight of the vessel nozzles, vessel flange, and the upper portion of the vessel barrel. Since, in fact, the reactor vessel shell, internals, fuel, and water in the vessel are not isolated from one another in terms of the response to a vertical load on the vessel flange, the entire weight of the vessel shell, vessel internals, fuel, and water was assumed for Part II to resist the impact of the falling head assembly. This weight was found to be 1650 kips.

c. The weight of the polar crane load block used in the analysis was reduced from an assumed value of 32 kips to an actual value of 16 kips. This reduced the weight of the dropped head package from 359 kips to 343 kips.

The reactor vessel supports for SNUPPS were evaluated to determine the maximum vertical displacement of the reactor vessel due to the postulated head drop accident. This was done using energy balance techniques, taking into consideration energy

losses at impact as determined by the change in velocity of the total system based on conservation of momentum principles.

The vessel supports are made up of a cooling box structure designed by Westinghouse that, in turn, is supported by a Bechtel design of structural steel framing partially embedded in the primary shield wall. The vertical stiffness of the Westinghouse-designed cooling box was found to be 4.07×10^5 kips/inch. This value is the resultant of several stiffnesses acting in series that were calculated separately along the height of the support. The separate calculations were necessary due to geometric changes in the cooling box which affected the bearing area of the vertical plates of which the box is composed.

The minimum moment of inertia along the height of the support was found to be 9,355 in.^{4}. This value was assumed constant throughout the unbraced length of 21.5 inches in order to find the minimum Euler buckling load of the cooling box. The buckling load ranges between 5.59×10^6 kips for pinned-pinned end conditions and 22.37×10^6 kips for fixed-fixed end conditions.

The minimum yield load of the cooling box was found by multiplying the cross-sectional area at the critical section (the same area used to determine the minimum moment of inertia) by the nominal yield strength of the steel. These values are 242.25 square inches and 50 ksi, respectively, and give a resulting yield load for each cooling box of 12.11×10^3 kips. A comparison of the yield load and minimum buckling load shows that the cooling box will yield before it buckles.

The yield displacement of the cooling box was found by dividing the yield load of 12.11×10^3 kips by the vertical stiffness, 4.07×10^5 kips/inch. The resulting displacement is 0.0298 inch. This defines the limit of elastic vertical deformation. Any further deformation was assumed to be perfectly plastic.

The Bechtel-designed portion of the reactor vessel supports has the following properties, as provided by Bechtel:

Stiffness:	34,880 kips/inch
Elastic Capacity:	7560 kips per support
Ductility Ratio:	10

From the above data, the yield displacement was found to be 0.22 inch with an ultimate deformation of 2.2 inches. As in the cooling box portion of the support, any deformation beyond the yield displacement was assumed to be perfectly plastic.

The stiffness of the reactor vessel upper barrel used in the support evaluation was revised to 1.550 x 10^6 kips/inch. This was the result of changes in the following parameters affecting the stiffness calculation. These values were taken from the vessel drawing specific to SNUPPS.

E = 28,000 ksi

R = 96.35 inches

r = 85.60 inches

A = 6,145 square inches

L = 111 inches

The parameter labels are defined in the reactor vessel nozzle evaluation of Part I.

The total system stiffness was found by adding the following stiffnesses in series:

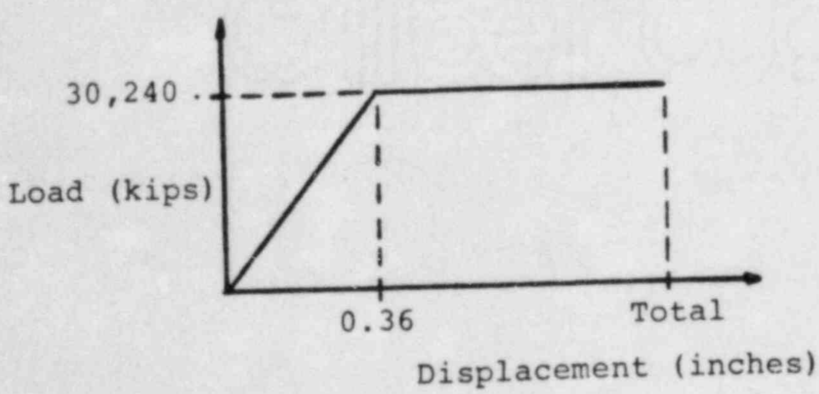
a. Reactor vessel inlet and outlet nozzle stiffnesses (two each, from Part I)

b. Cooling box and embedded structural steel stiffnesses (four each)

c. Reactor vessel upper barrel stiffness

The resulting total system stiffness is 85,270 kips/inch.

A bilinear load-displacement curve describing the system response was next developed. A comparison of the minimum yield loads of the cooling box and embedded structural steel showed that the embedded structural steel will yield first at 7560 kips per support or a total of 30,240 kips for all four supports. Dividing this yield load by the system stiffness gives a yield displacement of the total system of 0.36 inch. The following load-displacement curve can then be developed:



The area under the load-displacement curve is the internal resistance of the system to the impact load and system deformation caused by the postulated head drop accident. The area under the load-displacement curve, with X being the total displacement, is found as follows:

$$I = 30{,}240\ (X) - 0.5\ (30{,}240)\ 0.36$$

Where I = internal resistance

30,240 = minimum yield load of support system (kips)

0.36 = yield displacement of support system (inches)

The external work on the system is done by the falling head assembly and is the sum of the kinetic energy remaining after impact and the work required to deform the system vertically a distance X. The total kinetic energy at impact is modified to account for that which is lost by dissipation. The modification is by a factor determined by conservation of momentum and is given as follows:

$$\frac{343}{343 + 1650} = 0.172$$

Where 343 = weight of the falling head assembly (kips)

1650 = weight assumed to resist impact (kips)

The external work is given as follows:

$$E = 343\ (168)\ 0.172 + 343\ (X)$$

Where 168 = height of drop (inches)

Assuming the internal resistance equals the external work less the dissipated energy, the total system deformation was found by equating the expressions for I and E and then solving for X. The value of X was found to be 0.51 inch.

The distribution of the elastic deformation of the various components involved in supporting the reactor vessel is shown in Table C-2. These values were determined by multiplying the total system elastic deformation of 0.36 inch by the ratio of the stiffness of the specific component to the total system stiffness. The plastic deformation of 0.15 inch occurred entirely in the embedded structural steel portion of the vessel supports, since the minimum yield load assumed for the total

system was that of the embedded steel. The total deformation of the embedded steel was found to be 0.37 inch. This is less than the limit of 2.2 inches given previously.

After the maximum vessel displacement was determined, the routing of the essential auxiliary lines attached to each loop was examined to locate any possible interferences with civil structures, equipment, or pipe whip restraints. No interferences were found for the total vessel displacement indicated in Table C-2.

The results of the support evaluation for the postulated head drop accident show that the reactor vessel supports and, hence, the reactor vessel and nozzles, will displace vertically a maximum of 0.51 inch. Permanent deformation of the support system is 0.15 inch. The maximum displacement of the reactor vessel will not have an effect on the ability of the reactor coolant loop piping and essential auxiliary piping to circulate borated water to the core and remove residual heat. Therefore, core cooling capability and fuel cladding integrity are maintained.

App. C REFERENCES

1. Alexander, D. W., Shakely, R., and Dudek, D. F., "Reactor Vessel Head Drop Analysis," WCAP-9198, January, 1978.

2. Hunsaker, J. C. and Rightnire, B. G., Engineering Applications of Fluid Mechanics, page 183.

3. Hoemer, J. F., Fluid-Dynamic Drag, page 317.

4. Roark, R. J., Formulas For Stress and Strain - Fourth Edition, pages 340, 370, and 371.

TABLE C-1

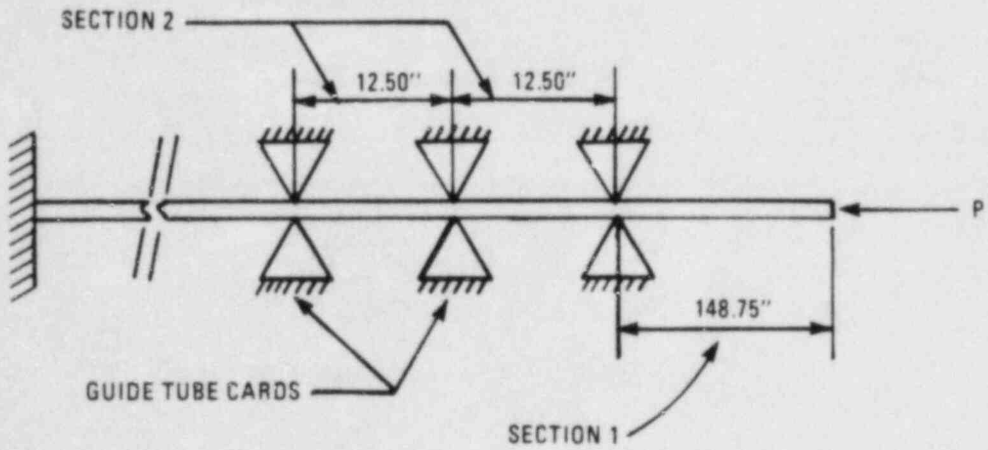
WEIGHT OF REACTOR VESSEL HEAD

Head	165,100
CRDM (full length) - (1500 lb/ mechanism; 53 mechanisms)	79,500
Rod position indicator coil stack	14,895
Seismic platform	11,100
Stud tensioner hoist	900
Dummy cans	848
Sling block platform	570
Head insulation	1,700
Lifting rig and vent shroud	37,000
Contingency	15,000
	326,663
Plus 10 percent (for load block)	32,666
Total	359,329 lbs

TABLE C-2

VERTICAL DEFORMATION OF REACTOR VESSEL AND SUPPORTS
DUE TO A POSTULATED HEAD DROP ACCIDENT

Supporting Component	Elastic Deformation (inches)	Plastic Deformation (inches)
Vessel Plus Nozzles	0.12	None
Westinghouse Cooling Box	0.02	None
Bechtel Embedded Framing	0.22	0.15
Subtotals	0.36	0.15
TOTAL DISPLACEMENT	0.51 inch	



SECTION 2 OCCURS A TOTAL OF 6 TIMES DOWN THE GUIDE TUBE

RADIAL CLEARANCE BETWEEN DRIVE ROD AND CARD = 0.325"
LARGEST DIAMETER THAT CAN PASS THROUGH GUIDE TUBE CARD - 2.4"

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FIGURE C-1
MODEL OF CRITICAL BUCKLING LOAD FOR THE DRIVE ROD

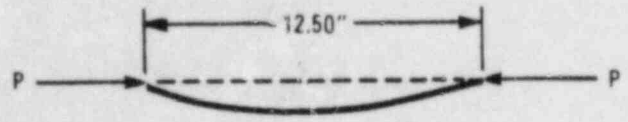

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FIGURE C-2
BUCKLING LOAD FOR SECTION 2

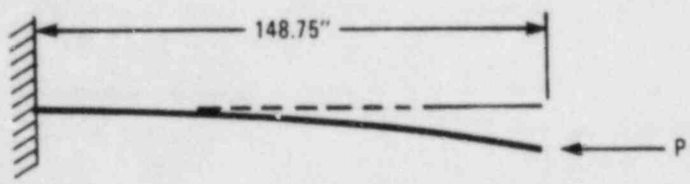

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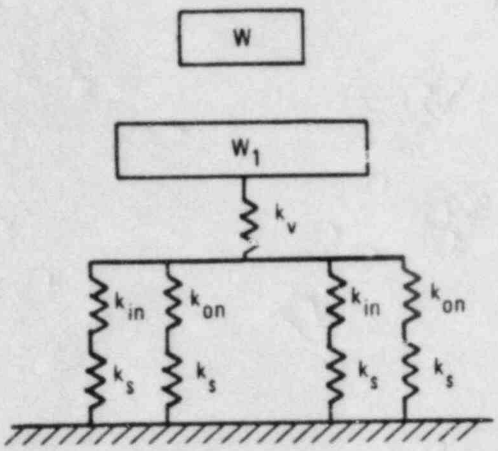
FIGURE C-3
BUCKLING LOAD FOR SECTION 1



W = WEIGHT OF THE UPPER PACKAGE, HEAD POLAR CRANE HOOKS, AND CABLE
W_1 = WEIGHT OF THE VESSEL FLANGE, NOZZLES, AND REGION IN BETWEEN
k_{in} = SPRING CONSTANT OF INLET NOZZLE REGION
k_{on} = SPRING CONSTANT OF OUTLET NOZZLE REGION
k_s = SPRING CONSTANT OF SUPPORTS
k_v = SPRING CONSTANT OF VESSEL AND FLANGE USING EQUIVALENT CYLINDER ANALYSIS

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FIGURE C-4
SPRING MASS SYSTEM OF THE HEAD UPPER PACKAGE AND REACTOR VESSEL

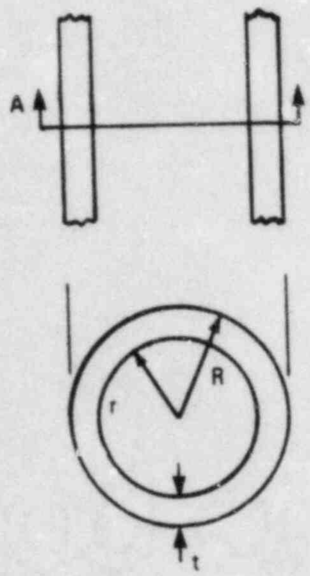

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FIGURE C-5
UPPER PORTION OF VESSEL

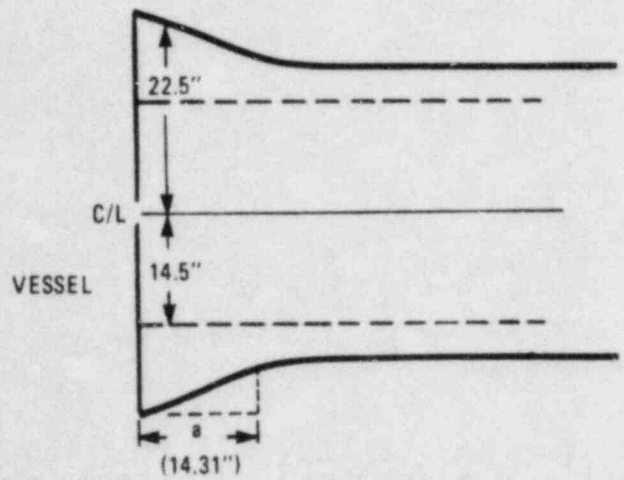

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FIGURE C-6
OUTLET NOZZLE

APPENDIX E
EVALUATION OF SNUPPS
REACTOR VESSEL HEAD AND INTERNALS LIFTING DEVICES

Introduction

The lifting devices for SNUPPS plant (Callaway Plant Unit No. 1 and Wolf Creek Generating Station Unit No. 1), which can be categorized as special lifting devices and which were evaluated in WCAP-10164, are:

1. Reactor vessel head lift rig
2. Reactor vessel internals lift rig
3. Load cell and load cell linkage

The reactor vessel head lift rig, the reactor vessel internals lift rig, load cell and load cell linkage were designed and built for Wolf Creek and Callaway circa 1975-76. These devices were designed to the requirements that the resulting stress in the load carrying members when subjected to the total combined lifting weight should not exceed the allowable stresses specified in the AISC code. A 125 percent load proof test was required on both devices followed by appropriate non-destructive testing. These items were not classified as nuclear safety components and requirements for formal documentation of design requirements and stress reports were not applied to the extent required for safety-related equipment. Thus, stress reports and design specifications were not formally documented. However, Westinghouse defined the design, fabrication and quality assurance requirements on detailed manufacturing drawings and purchase order documents.

Equipment Description

REACTOR VESSEL HEAD LIFT RIG

The reactor vessel head lift rig is a three legged carbon steel structure, approximately 48 feet high and 16 feet in diameter, weighing approximately 16,000 pounds. It is used to handle the assembled reactor vessel head.

The three vertical legs and Control Rod Drive Mechanism (CRDM) platform assembly are permanently attached to the reactor vessel head lifting lugs. The legs, clevis, and pins which are a part of the support for the seismic platform meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF Class I Supports. The tripod assembly is attached to the three vertical legs and is used when installing and removing the reactor vessel head. During plant operation, the sling assembly is removed and the three vertical legs and platform assembly remain attached to the reactor vessel head.

REACTOR VESSEL INTERNALS LIFT RIG

The internals lifting rig is a three-legged carbon and stainless steel structure, approximately 30 feet high and 14 feet in diameter weighing approximately 21,000 pounds. It is used to handle the upper and lower reactor vessel internals packages. It is attached to the main crane hook for all lifting, lowering and traversing operations. A load cell linkage is connected between the main crane hook and the rig to monitor loads during all operations. When not in use, the rig is stored on the upper internals storage stand.

The reactor vessel internals lift rig attaches to the internals package by means of three rotolock studs which engage three rotolock inserts located in the internals flange. These rotolock studs are manually operated from the internals lift rig platform using a handling tool which is an integral part of the rig. The studs are normally spring retracted upward and are depressed to engage the inserts. Rotating the mechanism locks it in both positions.

LOAD CELL AND LOAD CELL LINKAGE

The load cell is used to monitor the load during lifting and lowering the reactor vessel head or internals to ensure no excessive loadings are occurring. The unit is a load sensing clevis type, rated at 350,000 pounds.

This load cell is a part of the load cell linkage which is an assembly of pins, plates, and bolts that connect the polar crane main hook to the

lifting blocks of both the reactor vessel head and the internal lift rigs.

Evaluation Summary

Westinghouse performed an evaluation of the Wolf Creek and Callaway reactor vessel head and internal lift rigs, load cell and load cell linkage to determine the acceptability of these devices to meet the requirements of NUREG 0612. The evaluation consists of: (1) a comparison report of the ANSI N14.6 requirements and the requirements used in the design and manufacture of these devices; (2) a stress report in accordance with the design criteria of ANSI N14.6; and (3) a list of recommendations to enable these devices to demonstrate compliance with the intent of NUREG 0612 and ANSI N14.6.

The following paragraphs summarize the results of the evaluation.

Although no design specification was written for these devices, assembly drawings and manufacturing and purchase documents contain material requirements, and welding, non-destructive testing and coating requirements.

Stress reports and critical item lists have been prepared for each device.

Any repair to the devices is considered to be in the form of welding. Should pins, bolts, or other fasteners need repair, they should be replaced, in lieu of repair, in accordance with original or equivalent requirements for material and non-destructive testing.

High-strength materials are used in the devices. Although fracture toughness was not determined, material was selected based on its excellent fracture toughness characteristics.

Decontamination was not specifically addressed in the design of the devices. However, the design of many of the components (locking devices, pins, etc.) include features which would minimize decontamination efforts.

The Reactor Vessel Head and Internal Lift Rigs and load cell were proof tested upon completion with a load of approximately 1.25 times the design weight.

- Upon the completion of the test, all parts, particularly welds, were visually inspected for cracks or obvious deformation. Critical welds were magnetic particle inspected. In addition, the Westinghouse Quality Release verifies that the criteria for letters of compliance for materials and specifications required by the Westinghouse drawings and purchasing documents were satisfied.

It is obvious from their designs that these rigs are special lifting devices and can only be used for their intended purpose. Therefore, specific identification of the rig by marking the rig name and rated capacity is not required.

Liquid penetrant, magnetic particle, ultrasonic and radiograph inspections were performed on identified items. These were in accordance with ASTM specifications, ASME Code, Westinghouse process specifications or as noted on detailed drawings and provide similar results to the requirement of the ASME Code.

These special lifting devices are used during plant refueling which is approximately once per year. During plant operation these special lifting devices are inaccessable since they are permanently installed and/or remain in the containment. They cannot be removed from the containment unless they are disassembled and no known purposes exist for disassembly. Load testing to 150 percent of the total weight before each use would require special fixtures and is impractical to perform. An alternative check (visual) of critical welds and parts should be conducted at initial lift prior to moving to full lift and movement for these devices. Further, with the use of the load cell for the head and internals lift rig, all lifting and lowering is monitored at all times.

Application of the ANSI N14.6 criteria of these special lifting devices results in acceptable stress limits for tensile and shear stresses. Application of this criteria to all structural members subject to other types of loadings tend to result in oversimplified conservatism and with some stresses exceeding the accompanying allowable limits. However, when using the more appropriate criteria for those cases not addressed by the ANSI N14.6 criteria, the stresses are within the appropriate allowable limits.

The stress design factors required in ANSI N14.6 adequately address the NUREG 0612 issue of combined static and dynamic loads.

Westinghouse also provided recommended administrative controls (inspections, maintenance, and procedures) to assure continuing compliance with certain provisions.

Conclusions

The following conclusions result from this evaluation:

1. The ANSI N14.6 requirements for design, fabrication and quality assurance are generally in agreement with those used for these special lift devices.

2. The ANSI N14.6 criteria for stress limits associated with certain stress design factors for tensile and shear stresses are adequately satisfied.

3 These devices meet the ANSI N14.6 criteria for tensile and shear stresses and meet other appropriate criteria for loading conditions that result in combined and bearing stresses.

4. Administrative Controls

   a. Maintenance and inspection procedures will include a visual check of critical welds and parts during lifting to comply with ANSI N14.6 requirements for functional testing.

   b. Operating instructions and maintenance instructions will be reviewed to assure that they contain the requirements to address maintenance logs, repair and testing history, and damage incidents.

   c. Maintenance and repair procedures will refer, as much as possible, to requirements that were used in the original fabrication and will also define bolts, studs and nuts as non-repairable items.

d. To comply with provisions for acceptance testing and continuing compliance, the procedures for use of the reactor vessel and internals lift rigs will include the following:

1) Reactor Vessel Head Lift Rig:

A visual check of all welds will be performed prior to use and after reassembly of the spreader assembly, lifting lug, and upper lifting legs to the upper portion of the lift rig. At initial lift, the vessel head will be raised slightly above its support and held for 10 minutes. During this time, a visual inspection of the sling block lugs to the lifting block welds, and spreader lug to spreader arm weld will be performed. If no problems are apparent, the lift will continue while monitoring the load cell readout at all times.

2) Reactor Vessel Internals Lift Rig

Prior to use, a visual check will be performed to inspect the rig components and welds while on the storage stand for signs of cracks or deformation. A check of all bolted joints will be performed to ensure that they are tight and secure. After connection to the upper or lower internals, the assembly will be raised slightly off its support and held for 10 minutes. During this time, a visual inspection of the sling block lugs to the lifting block welds will be performed. If no problems are apparent, the lift will continue while monitoring the load cell readout at all times.

Based on the above, the SNUPPS reactor vessel head lift rig, reactor vessel internals lift rig and load cell linkage adequately comply with the guidelines of NUREG-0612 and ANSI N14.6.