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Integrity of the Reactor Vessel Support System for a Postulated Reactor Vessel Closure Head Drop Event

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Abstract

The integrity of reactor vessel support system of the Korean Standard Nuclear Power Plant (KSNPP) is investigated for a postulated reactor vessel closure head drop event. The closure head is disassembled from the reactor vessel during refueling process or general inspection of reactor vessel and internal structures, and carried to proposed location by the head lift rig. A postulated closure head drop event could be anticipated during closure head handling process. The drop event may cause an impact load on the reactor vessel and supporting system. The integrity of the supporting system is directly relevant to that of reactor vessel and reactor internals including fuels. Results derived by elastic impact analysis, linear and non-linear buckling analysis and elasto-plastic stress analysis of the supporting system implied that the integrity of the reactor vessel supporting system is intact for a postulated reactor vessel closure head drop event.

1. Introduction

The rector vessel support system of the Korean Standard Nuclear Power Plant, hereafter referred as KSNPP, consists of four vertical columns supporting each cold leg which has a vessel support flange designed to match with each vertical column. The main purpose of reactor vessel support system is to release the thermal expansion of reactor coolant system mainly caused by high temperature and pressure of reactor coolant. Another task is to ensure the integrity of reactor vessel and internal structures for seismic event and postulated pipe break events by confining the dynamic motion of reactor vessel. Figure 1 shows a brief configuration of the reactor vessel support system. The reactor vessel support system prevents the horizontal motion of the reactor vessel by allowing only small dimension of gaps at upper and lower part of the support column. Any additional restraints to the vertical and radial direction is not present on the other hands.

The reactor vessel closure head is separated from reactor vessel and transported to the proposed storage location using head lift rig for refueling process or general inspection. A postulated closure head drop event might be considered during these processes. Though this event may not be classified as a design base of the support system, the gross behavior of the support system should be closely monitored to estimate impacts on the safety or integrity of reactor vessel and internal structures including fuel assemblies. If this event happens, the closure head shall directly hit the reactor vessel and cause an impact load on the reactor coolant system. The impact load introduced by closure head drop event transmits a large amount of kinetic energy to reactor vessel and support system within very short time, and the gross failure or buckling of the support column could be expected. Since the postulated head drop is not classified as a design base event, the estimation of the possible failure or buckling of the support column due to closure head drop event shall be performed independent of other design basis.

In this study, the integrity of reactor vessel support system of KNSPP during a postulated closure head drop event was investigated on the bases of upper bounding assumptions.

2. Method of Analysis

2.1. Basic Assumption and Modeling

The main target of an accidental closure head drop analysis is to verify the integrity of reactor vessel support system under impact load. Some reasonable assumptions assuring enough conservatism to set up the upper boundary solution were established as follows.

1. Closure head is assumed to drop in air state. Any interference or structures confining the travel of the closure head is neglected.
2. An elastic impact concept is applied.
3. Since the reactor vessel and support system consists of a very thick cylinder and stiff columns, 3-D beam members are suitable to represent the structural characteristics.

The reactor vessel and support system were modeled by 3-D beam members on the base of above assumptions. Since the piping system which connects steam generator and reactor coolant pumps with reactor vessel is relatively flexible compared to reactor vessel and support system, other parts of reactor coolant system are converted to simple stiffness elements at the location where each piping system meets reactor vessel nozzle through equivalent stiffness analyses.

The reactor vessel model consists of 3-D beams and lumped mass point at the center of gravity location. All masses of internal structures, fuels, control element driving mechanism (CEDM) and water are considered to get the most conservative responses. Each support model consists of 3-D beams and 8 mass points which are evenly distributed to monitor the response of each support column. Typical masses considered for the KNSPP are listed in Table 1. Typical analysis model for impact analyses is shown in Figure 2.

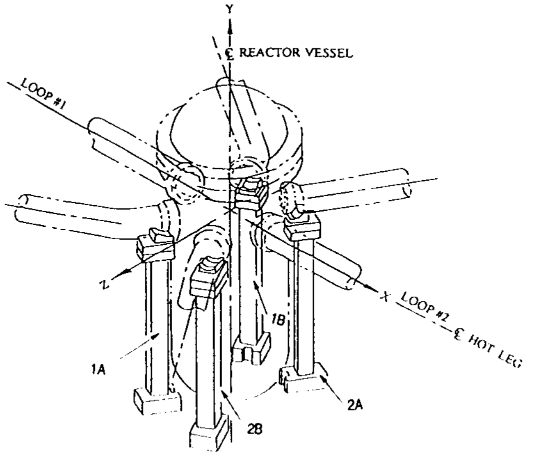

Fig. 1. Reactor Vessel and Supporting System

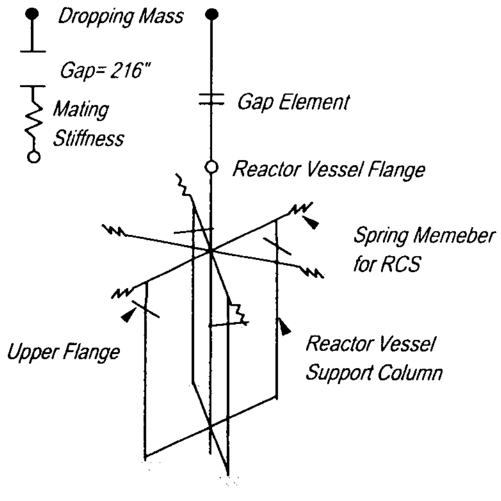

Fig. 2. Typical Impact Analysis Model

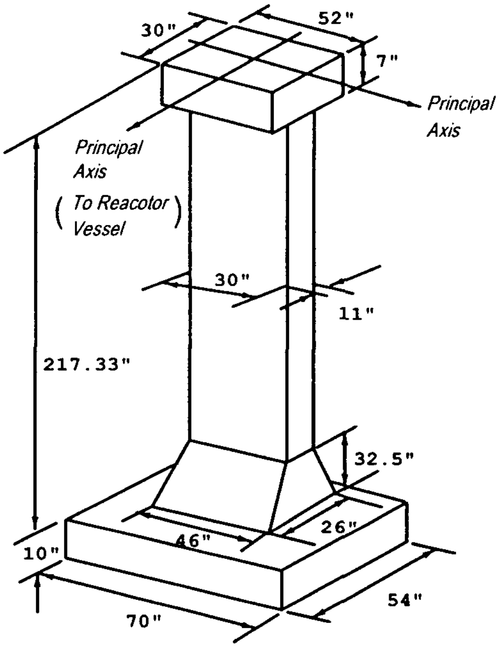

Fig. 3. Typical Dimension of the Reactor Vessel Support Column

Table 1. Typical Mass Distribution

Item	Weight (lb)
Reactor vessel with internals and water	1496220
Vessel support column with inlet nozzle	50349
Dropping Mass (Closure Head & Attachments)	193100
Outlet Nozzle	7151

Table 2. Typical Material Properties(3)

Property	Value
Elastic modulus (psi) at 120°F	27.53E6
Yield strength (psi)	50E3
Minimum tensile strength (psi)	80E3

2.2. Buckling Analysis of Support Column

Since basic configuration of each support column could be considered as a long slender beam as shown in Figure 3, the estimation of limiting load to prevent the possibility of linear or non-linear buckling event due to an impulsive load shall precede to determine the stability of each support column. Each support column has a rectangular cross section and one of its principal axis is aligned to the radial direction of reactor vessel. The thickness toward radial direction is designed to be flexible to release the thermal expansion of the reactor vessel, whereas the circumferential dimension is devised to support the dynamic motion of reactor vessel. Thus, the stiffness toward radial direction is more flexible than circumferential one.

One of the most simple and conservative method to forecast the buckling load is to apply the Euler's equation.(1)(2) Since the upper flange of each support column is assembled with cold leg nozzle flange providing a rigid boundary, the minimum buckling load by Euler's equation is given by fixed-fixed boundary condition.

The material used for typical support column is specified as SA-508 class 2 or 3 equivalent. Typical material properties are listed in Table 2. If the buckling capacity of a column is very high enough to cause yielding before buckling, the elastic limit of a support column shall represent the buckling limit of a support column within elastic range. Another method to review the elastic buckling behavior of a structure is to apply a small amount of imperfection to a column, then it triggers a non-linear buckling of the column to find out the snap through effect caused by the change of bending stiffness or geometric imperfection. Buckling analyses were performed using ANSYS Version 5.2(4) on the HP-Apollo 9000-735.

2.3. Dynamic Analysis of the Closure Head Drop Event

Two cases of impact event were assumed to find out an upper bound load. The first case is a concentric drop of closure head from the 18 feet elevation without rotation. The other case considered the possible impact of closure head with oblique angle due to rotation. When the oblique impact happens, the distribution of dropping mass and mating stiffness of reactor vessel shall be different case by case. To simulate the oblique impact, a rigid member connecting reactor vessel center and arbitrary location on the reactor vessel flange is added to the basic model. Total dropping mass is applied at the arbitrary flange location to envelop an extreme case. Since the mating area between closure head and reactor vessel shall be reduced and interfered by other internal structure or building structure, half of the stiffness of reactor vessel member above the nozzle location were used as mating stiffness to get conservative results. The current model introduced the fundamental vertical frequency around at 28Hz. Then the damping factor of 3%(4)(5)(6)(7) was applied below the frequency range of 33Hz using stiffness proportional damping. Since the gap element requires very small integration time step, typical time interval of 1E-4 second was used to pick up the response of the structure. Impact analyses were performed using ANSYS Version 5.2(4) on the HP-Apollo 9000-735.

3. Results and Discussion

Table 3 shows a collection of static analysis results including non-linear buckling and elasto-plastic analysis. The length of column used for Euler's load was derived on basis of minimum cross section and an amount of 5%(2)(4) imperfection were assumed for non-linear buckling analysis. The result of each static analysis in Table 3 shows that the minimum static limit of a support column could be defined through elasto-plastic analysis. In other words, the buckling phenomenon may not occur before the failure of support column and the limiting load for a support column shall be determined on the base of elasto-plastic analysis result. If the impact load due to closure head drop does not pass over this limit so much, each support column might be intact during drop event.

Figures 4 through 6 show time histories of displacement and reaction load of reactor vessel and support column for concentric impact case. The load developed at mating surface, i. e., reactor vessel flange, shows a sharp peak and vanishes out quickly as shown in Figure 5. Though the impact load applied vanishes quickly, responses at the support column are magnified due to the motion of reactor vessel side. The reaction load of each support column shown in Figure 6 could be followed by the motion of reactor vessel shown in Figure 4. The first peak shown in Figure 6 might come from the direct transmission of an impact from reactor vessel flange and the rest of reaction load should be developed by the

Table 3. Results of Static Analysis

Method	Limit load (lb)
Classical Euler's Equation1)	2.606E7
Linear buckling analysis2)	3.830E7
Nonlinear buckling analysis2)	4.813E7
Elasto-plastic analysis2)	1.721E7
Method of critical cross section3)	2.640E7

1) Minimum solution

2) Run by ANSYSElastic-perfectly plastic material model used.

3) Use only minimum cross section :
Min. Tensile Strength x Min. Cross Section

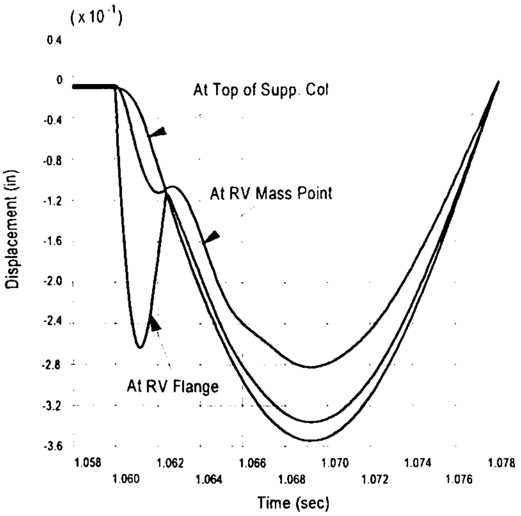

Fig. 4. Displacement Time Histories for Concentric Drop Event

motion of reactor vessel. The maximum axial load developed due to the concentric impact case is estimated as 1.20E7 pounds, which is less than the minimum static load defined through Table 3. Thus it is believed that resultant stress of each support column built during concentric impact resides within the elastic range.

Figures 7 through 10 show results of an oblique impact case. Since the closure head was assumed to hit certain edge of reactor vessel flange, non-symmetric responses could be expected as shown in Figure 10. The mating location was assumed at above of the supporting column 1A as designated in Figure 1 to get the most conservative response of a support column. Though the displacement developed by oblique impact case is larger than that of concentric one, the axial loads for reactor vessel entry(see Fig. 5 and 9) are decreased compared to concentric case.

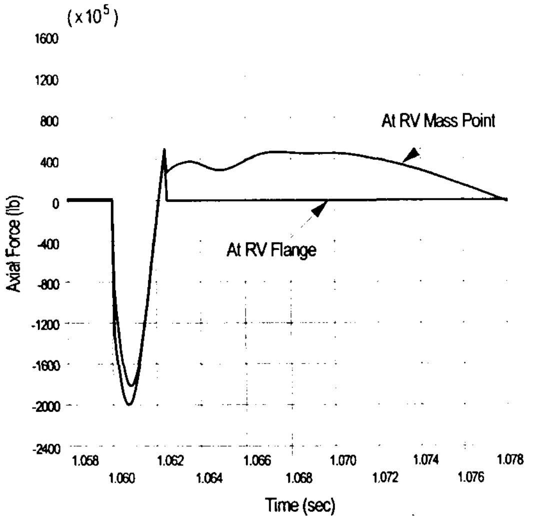

Fig. 5. Axial-Force Time Histories for Concentric Drop Event

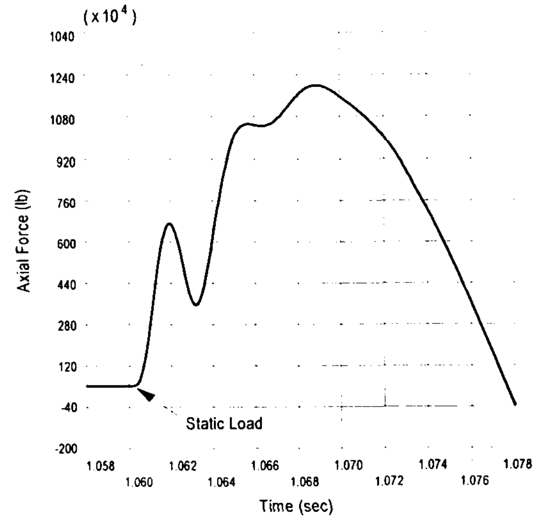

Fig. 6. Axial-Force Time History of Support Column for Concentric Drop Event

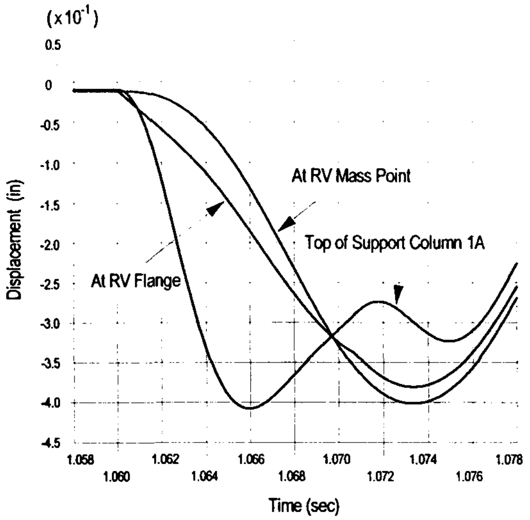

Fig. 7. Displacement Time Histories for Oblique Drop Event

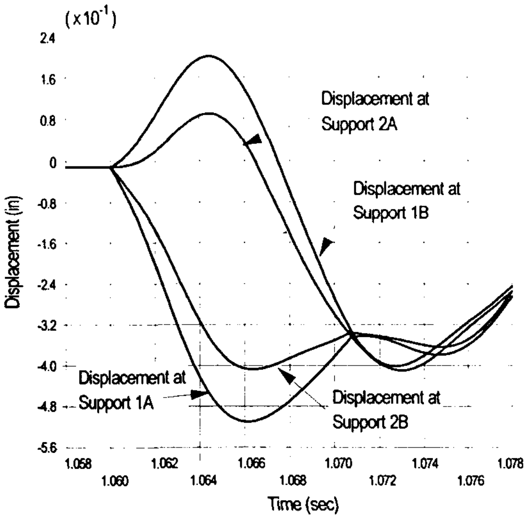

Fig. 8. Displacement Time Histories of Support Columns for Obeique Drop Event

Since the oblique impact causes non-symmetric loading to each support column as shown in Figures 8 and 10, the interaction between each column introduces crossing interference for each responses and results oscillating responses. The compressive axial forces, represented as a tensile reaction in Figure 10, are introduced at 1A and 2B support column and tensile forces are developed at 2B and 1B side in turn. Since the oblique impact introduces non-symmetric loads to each column, the surplus bending moments and shear forces may increase resultant stress occurring at a column section. But the maximum stress caused by the bending moment is calculated less than 5% of gross axial stress caused by an axial force. The maximum axial load of 1.77E7 pounds was developed at support column 1A where the closure head meets reactor vessel. Since this value is slightly higher than the elastic limit of a support column, a small scale plastic zone may be expected. However, since the load defined by critical cross section theory is still higher than this load, the gross failure of the support column is prevented.

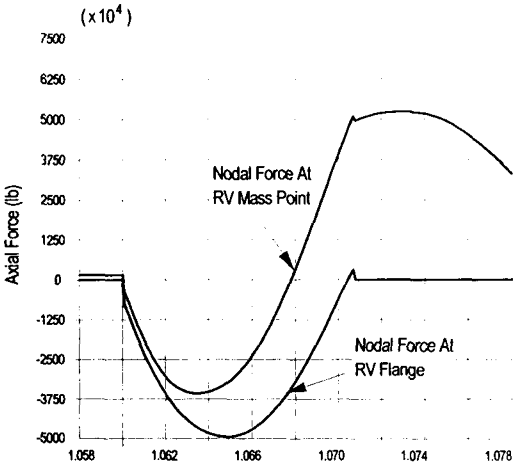

Fig. 9. Axial-Force Time Histories for Oblique Drop Event

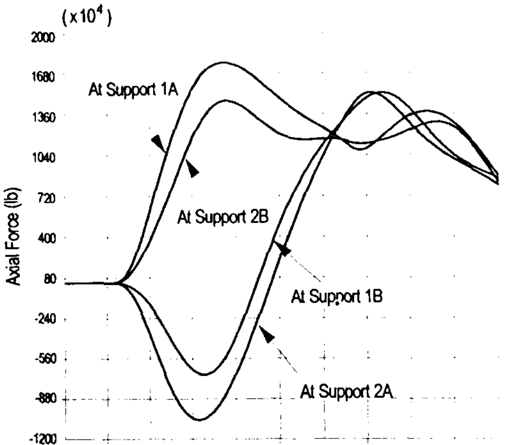

Fig. 10. Axial-Force Time Histories of Support Columns for Oblique Drop Event

4. Conclusions

The integrity of a reactor vessel support column of the Korean Standard Nuclear Power Plant was investigated on the base of a postulated closure head drop event. Two cases of head drop event were considered to estimate the integrity of the support column. The linear and non-linear buckling analysis showed that the buckling load resides beyond the elastic limit of the support column. And the resultant stresses caused by concentric head drop event remained in the elastic range. Though the oblique impact might cause certain plastic zone over a support column, enough margin exits to cover the gross failure of the support column.

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