Effect of Pump Seizure on the Qualification of Primary Heat Transport System for SSE

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ABSTRACT In nuclear Power Plants, the Primary Heat Transport System (PHTS) is designed according to ASME Section 3 Division 1. The PHTS together with Primary Coolant Pumps (PCPs) is designed so that the integrity under Safe Shutdown Earthquake (SSE) is maintained, but the operability under SSE is not ensured. There might be a common mode seizure of PCPs under SSE as they are not designed for operability under SSE. The pump seizure has two effects, both of which may affect the integrity of PHT system. The first effect is development of short term high magnitude pressure transient and the second effect is high loads at the pump support due to high inertia of the flywheel attached to the pump shaft. This paper presents and discusses the abovementioned two effects. The study shows that the peak pressure reached during the transient is much higher than the operating pressure. The system integrity may be affected if the design pressure is significantly below the peak pressure or if the pump support is not designed to resist the imposed torque.

INTRODUCTION

In nuclear Power Plants, the Primary Heat Transport System (PHTS) is designed according to ASME Section 3 Division 1. This requires that the system integrity should be maintained under SSE. Although the PHTS together with PCPs are designed so that SSE requirement is met, the PCPs operability under SSE is not ensured. Consequently PCPs may seize under SSE. This seizure may be a common mode failure for all PCPs. This has two effects.

(i) It introduces high magnitude pressure surge in the PHTS.
(ii) It introduces significantly higher load on the pump support.

Indian PHWRs of 220 MWe have two sets of two pumps, each set is on either side of the PHTS loop. In case of a SSE all four pumps may seize due to common mode seizure. The paper presents and discusses the pressure transients introduced due to pump shaft seizure. The analysis is carried out using thermal hydraulic analysis code RELAP4/Mod6. The load experienced by pump support is also discussed.

Description of PHTS of 220 MWe PHWRs

Indian PHWRs have horizontal core geometry. These are pressure tube type reactors where coolant flows through the pressure tube in the core. The PHT system of the 220 MWe PHWRs
(Fig. 1) consists of a number of parallel channels in a figure of eight loop configuration, with one pair of Steam Generators (SGs) and PHT pumps on each leg of the loop. Out of the total 306 fuel channels, flow through 153 channels is from south to north and the flow direction in the remaining 153 channels is from north to south. The inlet side of these channels is connected to the Reactor Inlet Header (RIH) and the outlet side is connected to the Reactor Outlet Header (ROH) through feeder pipes. The coolant flows through the fuel channels (pressure tubes), each of which contains twelve fuel bundles. Each pressure tube is surrounded by a calandria tube, which in turn is surrounded by the relatively cold moderator. The annular space between the pressure tube and the calandria tube is filled with carbon dioxide. The primary side of the Steam Generators (SG) consists of U tubes and the secondary side consists of the riser part, the steam drum and the downcomer. The feed to the secondary side of SG is supplied with the help of Main Boiler Feed Pump (MBFP) and on failure of these, when required the Auxiliary Boiler feed Pump (ABFP) comes in. The Reheat Condensate Recirculation Flow (RHCRF) is also fed in the secondary side of the SG. The PHT system pressure at 8.6 MPa is maintained with Feed-Bleed system. The Feed injection takes place at the ROH with the help of the Pressurising Pump which in turn is connected to storage tank of 6.0 tonne heavy water capacity. The Pressure and the level of the secondary side of SGs are controlled with Boiler Pressure Controller (BPC) and Boiler Level Controller (BLC).

In this analysis the PHT system is modelled with control volumes, junctions, core heat slabs, steam generator heat slabs, valves, fill and leak options (Fig. 2). The analysis is carried out with thermal-hydraulic analysis computer code RELAP4/MOD6 [1].

Description of PCP Support
The PCP is hung with aid of two variable support spring hangers and provided with sliding support at the bottom. The hangers are taking 80% of the weight and sliding support is taking the rest. To mitigate large displacements of pump-motor unit during an earthquake, horizontal snubbers are provided. Two horizontal snubbers are provided at the motor location. Two more snubbers are provided at the pump bowl location.

RELAP4/MOD6 Code
For analysis with the RELAP4/MOD6 code, the thermofluid systems are divided into a number of elements, usually called fluid control and nodal volumes. The fluid control volumes are connected by nodal volumes called junctions. Junctions are three general types, normal, leak and fill junctions, and they provide the means for the transfer of fluid into and out of the fluid volumes. This is a one dimensional code where the set of partial differential equations for (1) mass, (2) energy, and (3) momentum for each control volume in the system are solved simultaneously by conversion into control volume form and applying numerical techniques. The equations are as follows:

\[
\frac{\partial p}{\partial t} = -\frac{\partial W}{\partial x} \quad 1
\]

\[
A \frac{\partial (\rho e)}{\partial t} = -\frac{\partial}{\partial x} \left[ W \left( h + \frac{v^2}{2} + \phi \right) \right] + q_w \frac{\partial A_w}{\partial x} \quad 2
\]
\[
\frac{\partial (\rho v)}{\partial t} = \frac{\partial (\rho v W)}{\partial x} - A \frac{\partial P}{\partial x} - \rho g A \frac{\partial Z}{\partial x} - \frac{\partial F_k}{\partial x}
\]

For the property calculations, the thermodynamic data for water are obtained from the steam table which is included in the code. A heat conductor model is used to account for heat transfer to and from the fluid in a given control volume. The heat conduction equation (4) is solved using semi-implicit numerical technique.

\[
\rho C_p \frac{\partial T}{\partial t} = Q + k \frac{\partial^2 T}{\partial x^2}
\]

Wilson bubble rise velocity correlation is used to compute a bubble rise velocity knowing the mixture void fraction and volume pressure. This model assumes a linear increase in bubble density as a function of normalised height within a two-phase mixture equation (5).

\[
\rho_{bh} = a \left( Z / Z_m \right) + b
\]

Here, a and b are functions of rate of steam entrainment, volume of two-phase mixture and bubble velocity gradient.

Different heat transfer correlations are used for different heat transfer regimes. Component models that describe the behaviour of pumps, valves, steam separators etc. are used to describe various components of reactor systems. Control systems, in PHT system, are modelled using the different types of fill options that are defined through input data. Complete description of the code RELAP4/MOD6 is available in reference 1.

**POSTULATED SCENARIO**

1. The pumps seize at 0 s due to SSE. The seizure time is 1 s.
2. Reactor trips following pump trip on overload with 0.85 second delay.
3. Relief valve opens on PHTS pressure of 9.5 MPa with 0.5 second delay (Which includes instrumentation response time).

**RESULTS AND DISCUSSION**

Fig. 3 presents the pressure transient in the Reactor Outlet Header (ROH). The pressure steadily rises for about 4.8 s following pump seizure. It reaches a peak value of about 15.3 MPa from about 8.6 MPa. The reactor trips at 0.85 second. Relief Valves on the primary side open at 1 s. The pressure falls monotonically to 12.3 MPa when pressure oscillations are observed. The pressure falls below its normal operating value at 37 s.

Initial rise in the pressure is due to a mismatch between heat transferred to PHT system in the core and heat removed in the SGs and heat removed by relief valves. The mismatch occurs because the common mode seizure of all the pumps reduces the flow (Fig. 4) of the system significantly. This results in sudden decrease in heat transfer to secondary in the SGs as this process is dominantly temperature and heat transfer coefficient controlled process. The process in the core is dominantly heat flux and heat transfer coefficient controlled. Although reactor trips in about 1 s, the stored heat of the fuel continues to flow in the system. After about 4.8 s, the combined effect of relief valves, reactor trip and SGs is able to remove sufficient heat to bring down the pressure.
Calculation of Torque
The pump shaft seizure time is assumed to be 1 second. A typical pump case is taken whose starting torque is 36,000 Nm, rated speed 1481 rpm, operating torque 18,000 Nm and total moment of inertia including flywheel is 934 kg m². Then the resisting torque because of shaft seizure would be 180,161 Nm.

If the design pressure of the system is significantly below the peak pressure reached the system integrity may be affected. Similarly if pump support is not designed for pump seizure resisting torque then also the system integrity may be affected.

NOMENCLATURE

A = flow area
A_w = wall area
C_p = specific heat
e = total fluid specific energy
F_k = frictional force
g = gravitational acceleration
h = enthalpy
K = thermal conductivity
P = pressure
Q = volumetric heat generation rate
q_w = heat flux at wall
T = Temperature
v = velocity
W = mass flow rate
Z = control volume height
Z_m = mixture height in a volume
ρ = density
ρ_gb = Partial bubble density within the mixture
φ = gravity potential

REFERENCE:

FIG. 1 SCHEMATIC DIAGRAM OF PHT SYSTEM

NRIH  North Reactor Inlet Header
SRIH  South Reactor Inlet Header
NROH  North Reactor Outlet Header
SROH  South Reactor Outlet Header
PCP   Primary Coolant Pump
SG    Steam Generator
ABFP  Auxiliary Boiler Feed Pump
MBFP  Main Boiler Feed Pump
RHRDF Reheat Condensate Recirculate Flow
CSDVs Condensate Steam Dump Valves
ASDV8 Atmospheric Steam Discharge Valves

FIG. 2 NODALISATION SCHEME
FIG. 3 REACTOR OUTLET HEADER PRESSURE

FIG. 4 MASS FLOW (Jn. 73)