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STC PATON

SSTC NRS — STATE ENTERPRISE STATE SCIENTIFIC AND TECHNICAL CENTER FOR NUCLEAR AND RADIATION SAFETY

Developing Multi-step Approach of Nuclear Power Plant Equipment Integrity Assessment on Example of PSV (Pressurizer Safety Valve) Structural Analysis during Accident Scenario via System Coupling Simulation (Fluid Structure Interaction Transient Analysis vs Relap5 Code)

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MOTIVATION

The computationally intensive development made it possible to create (and implement) an innovative, comprehensive algorithm of safety assessment Nuclear Power Plant (NPP).

TARGET 1: To develop the Coupling System (Thermal-Hydraulic & Structural) basic model of Reactor Coolant System for further ability to pre-simulate any chosen accident scenario with Automatic Data Transfer between coupled CAE environments;

TARGET 2: To present a multi-step algorithm of equipment integrity assessment through an example aiming to obtain «realistic» Stress Strain State of Pressurizer Safety Valve.

CHALLENGES:
- FSI Simulation
- Realistic kinematic boundary condition on Valve supports for accident transients
- Rapid Thermal Changing Issue (The Temperature Differential is about 300°C)
- Rapid Pressure Changing Issue (The Pressure Differential is about 16MPa)

LIST OF INITIATING EVENTS (accidents) to be considered according to SAFETY GUIDELINES

...
3.2 Rupture of the line connecting the pressurizer and safety valve
3.3 Inadvertent opening of one pressurizer safety valve
3.4 Leaks from the primary to the secondary side of the steam generator
...

IAEA International Atomic Energy Agency

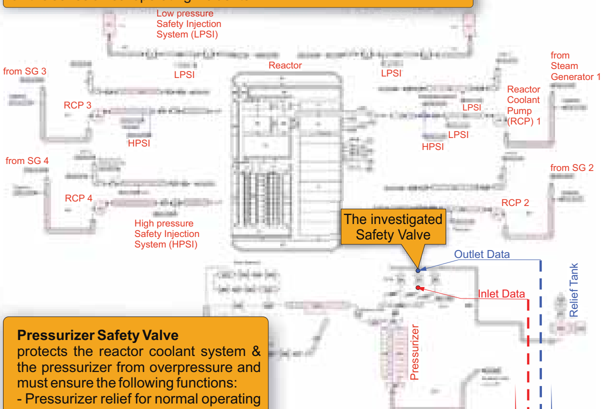
Stress Strain State Solution of Reactor Coolant System FE Model for Postulated Transient Accident Scenario

Termal Hydraulic model of Reactor Coolant System

How to validate?

The only methods for validating the Thermal-Hydraulic (TH) model are the computational analysis of transients (anticipated operational occurrences, emergencies, etc.) that occurred during the operation or testing of the Reactor Facility (RF) and its comparison with recorded data (obtained by measuring equipment of the NPP). The developed TH model was validated on the series of real operating incidents.

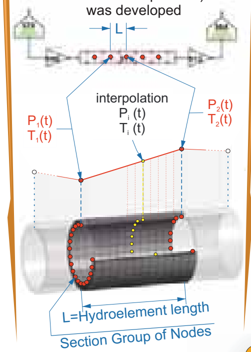
RELAP5 (Reactor Excursion and Leak Analysis Program)



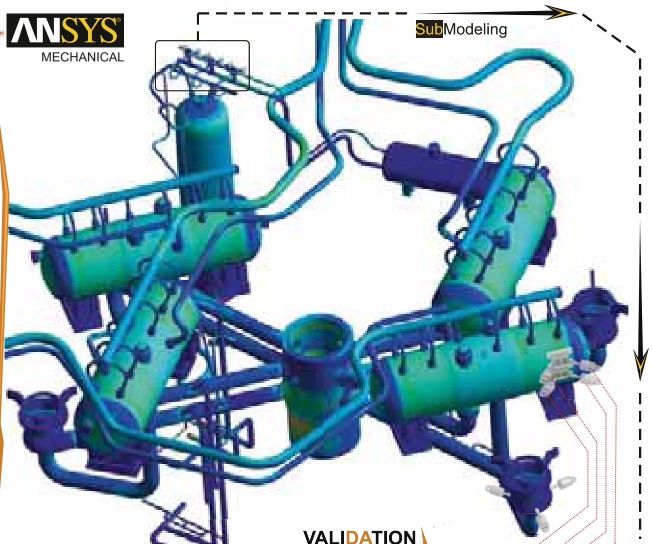
Solution Data Transfer Features:

ANSYS MECHANICAL

FE model of the Reactor Coolant System was designed divided into **Mesh Sections** relevant to the length of TH model hydro elements. Considering the challenge of large DATA sets needed to transfer automatically - **APDL Script** (based on linear interpolation) was developed

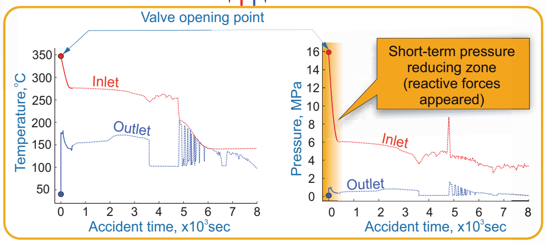


SubModeling

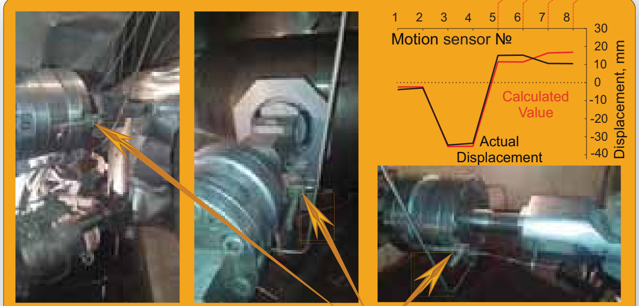


Pressurizer Safety Valve

protects the reactor coolant system & the pressurizer from overpressure and must ensure the following functions:
- Pressurizer relief for normal operating transients;
- Hot and cold overpressure protection on liquid, steam, gas and steam-water mixture;
- Depressurization of reactor coolant system following an accident.

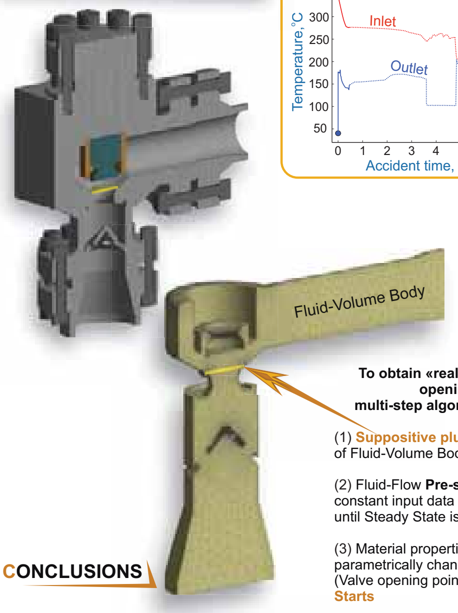


VALIDATION



The model has been validated using DATA SET of 53 Motion Sensors installed on Hydraulic Snubbers measured during the heating transient.

INITIAL STATE		THE END-STATE	
Primary Side:	P=3.5 MPa; T=90°C	Primary Side:	P=15.0 MPa; T=250°C
Secondary Side:	P=3.0 MPa; T=70°C	Secondary Side:	P = 6.0 MPa; T=200°C



ANSYS FLUENT

FSI

One way Fluid Structure Interaction Method

ANSYS MECHANICAL

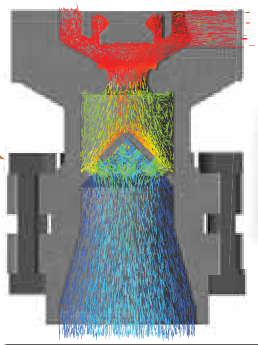
To obtain «realistic» short-term valve-opening simulation, multi-step algorithm was implemented:

(1) Suppositive plug was generated as a part of Fluid-Volume Body

(2) Fluid-Flow **Pre-simulation** starts with constant input data (normal operating mode) until Steady State is obtained

(3) Material properties from solid - to fluid parametrically change for Suppositive plug (Valve opening point) and Transient Process Starts

Transient Thermal State



RESULTS

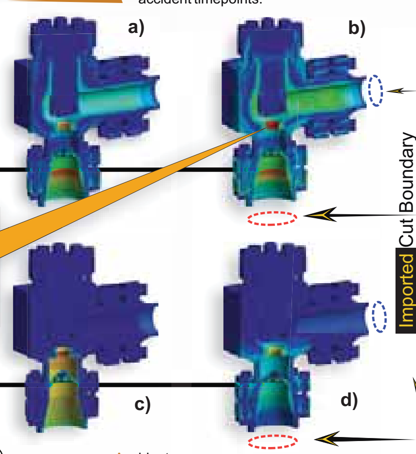
Stress Strain State for chosen accident timepoints:

COMPARATIVE ANALYSIS

Fig. a,b
Relap5 + FSI Method

Relap5+FSI System Coupling Method allows capturing the most **significant** local stress region principally caused by the dynamic evolution of thermal conditions

Fig. c,d
Relap5 + Transient Structural Analysis



20 sec — Accident Time — 180 sec

CONCLUSIONS

A Multi-Step Approach of Nuclear Power Plant (NPP) equipment integrity assessment was presented in two General Steps:

Step 1
- The Coupled System (Thermal-Hydraulic & Structural) basic model of the Reactor Coolant System (RCS) has been developed and assembled through an automatic data transfer application. The model was successfully validated using the values of parameters measured during the heating transient.
- The developed Basic model possesses a wide range of capacities for further, more accurate analysis of RCS facilities (for example: Steam Generator, Pressurizer, Relief Tank, etc):
  1) Analysis of RCS transient behavior for choosing the **most Adverse Accident Scenario** from International Safety Guidelines for RCS equipment supposed to be investigated
  2) Input Data for further submodeling Structural Analysis (realistic kinematic time-changing boundary conditions)
  3) Input Data for further detailed TH analysis

Step 2
- The FSI (Fluid Structure Interaction) Method was implemented for PSV (Pressurizer Safety Valve) Structural Analysis during Accident Scenario
- Termal-Hydraulic Input Data was taken from STEP 1 stage
- Imported Cut Boundary Constraint was applied to the investigated valve supports

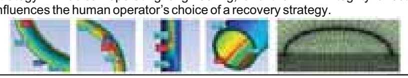
Comparative Solution analysis performed for developed Valve model in the most adverse points of accident time

The studies reveal the importance of detailed simulation with Fluid-Structure Interaction Method. The current simulation allows capturing the most **Significant Stress Zone** principally caused by dynamic evolution of thermal conditions.

Future Steps

Following the «chasing» of accurate numerical predictions and described the above algorithm, we are currently working on the **Integrity assessment** of Steam Generator Heat Exchange Tubes (HET) under Total Station Blackout Accident with feedwater restoration into drained/partially drained steam generator.

Challenge 1: Control of SG feeding is one of the most important recovery strategy in nuclear operating engineering, and the HET integrity directly influences the human operator's choice of a recovery strategy.



Challenge 2: Existing Cracks & Sludge