Typical Technology of Mechanics on Gen-III Passive NPPs and Gen-IV Advanced Supercritical Light Water Reactors

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Abstract. Technical requirements for Gen-III advanced nuclear power plants, which take passive reactors as the main body, were originally brought forward in American "Advanced Light Water Reactor Utility Requirement Document" (ALWR-URD) in early 1990's. The primary characteristic of passive nuclear power plant is large amount of simplification to the original active safety systems, replacing or supplementing them with passive safety systems, which enhances safety and economy. However, the replacement of active safety systems by passive safety systems also brings about some mechanics that compel attention, typically, such as load-carrying capability evaluation for steel containment, in-vessel retention (IVR) of molten core debris, seismic design without OBE, thermo-hydraulic issues concerning with coupling between two-phase fluid and solid, etc.

At the beginning of this century, six typical Gen-IV advanced reactor types (Sodium Cooled Fast Reactor, Supercritical Water-Cooled Reactor, etc.) were put forward. Among these types of reactors, Supercritical Water-Cooled Reactor adopts supercritical water as coolant and operates above the thermodynamic critical point of water by increasing temperature and pressure of the coolant, which makes the plant economic and efficient. However, this type of reactor also brings about some mechanical difficulties (e.g. pressure fluctuation caused by the supercritical fluid in the core, creep of materials working at high temperature, etc.) for the design of facility and components.

In this paper, the issues mentioned above are outlined for further consideration.

1. INTRODUCTION

American Electric Power has been devoted to establishing a technical foundation for the design of next generation of light water reactors (LWR) since 1980's. The Electric Power Research Institute (EPRI) has developed a comprehensive set of design requirements for the advanced LWR in the form of *Advanced Light Water Reactor-Utility Requirements document* (ALWR-URD), which defines the technical basis for innovated and standardized future LWR designs. ALWR Program policy statements are as follows: simplification, design margin, human factors, ALWR safety, ALWR design basis versus safety margin, regulatory stabilization, plant standardization, use of proven technology, maintainability, constructability, quality assurance, ALWR economics, ALWR sabotage protection, and ALWR good neighbor. Here the reason why simplification is put in the first place is that unnecessary complexity is considered to be a root cause of a wide range of problems in existing plants, as stated by ALWR Committee.

Therefore, guided by the URD and its policy, the plants built in last 90's were designed following the principle of simplification and some active safety systems of original plants were replaced or supplemented by passive safety systems relying on simple physical laws. Compared with traditional PWR plants, the numbers of valves, pumps, pipelines, cables, dampers in these plants reduced 50%, 35%, 80%, 70%, and 80%, respectively, the volume of seismic structures reduced 45%, while nuclear steam supply system remained utilizing proven technology[2]. So the design concept of ALWR is completely renovated compared with traditional Gen-II PWR, and ALWR plant may be called as Gen-III passive advanced plant. Simplification and passive design bring forward some new issues of mechanics, such as load-carrying capability evaluation for steel containment, in-vessel retention of molten core debris (IVR), seismic design without OBE, thermo-hydraulic issues concerning coupling between two-phase fluid and solid, etc.

Discussions on Gen-IV advanced nuclear power plant arose in the beginning of this century. The design concept of Gen-IV NPP focuses on renovation of reactor itself. For example, in a Supercritical Water-Cooled Reactor (SCWR), the temperature of coolant is increased above the thermodynamic critical point of water by the core and steam generators are canceled, which greatly increases thermal efficiency. However, besides the in-core thermo-hydraulic and fuel design problems, mechanical and material problems (e.g. pressure fluctuation with shock wave shape caused by the supercritical fluid in the core, creep of materials working under high temperature, etc.) are also main obstacles that prevent the concept from coming true.

In this paper, the issues mentioned above are outlined for further consideration.

2. MAIN MECHANICS FOR THE DESIGN OF GEN-III PASSIVE ADVANCED NPP

2.1. Main Features of Design

The main features of Gen-III passive advanced NPPs are simplification and passive design concept, which makes the plants safe and economic. As a typical example, the containment cooling system of AP1000 has a double-layered containment structure: the inner layer is steel containment and the outer, concrete structure (Fig. 2.1-1). As the third barrier of the plant, the steel containment serves to prevent radioactivity from releasing out to the atmosphere. And in the event of large LOCA and core meltdown accident, water of passive containment cooling water storage tank, which is incorporated into the shield building structure above the containment vessel, streams down the outer of the steel containment so that the steel containment vessel is cooled and steam inside the vessel wall is condensed to water, which decreases steam pressure and temperature in the containment and mitigates accident sequence [3].

2.2. Pressure Load-Carrying Capability of Steel Containment Vessel

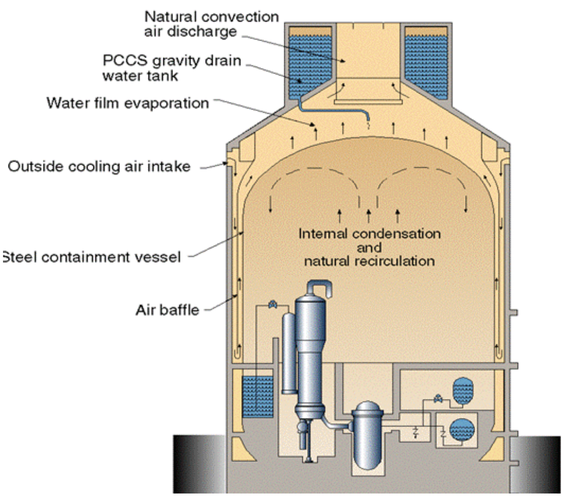


Fig. 2.2.1. Passive Containment Cooling System

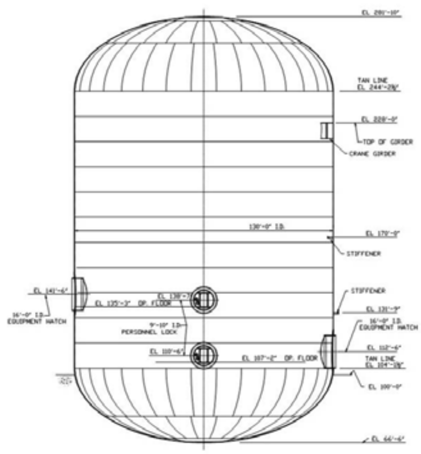
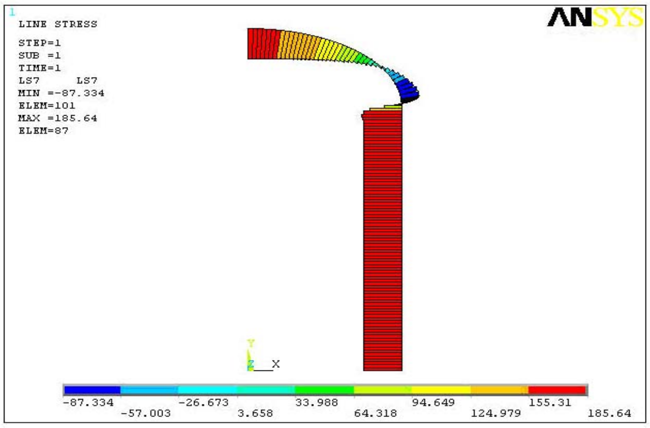
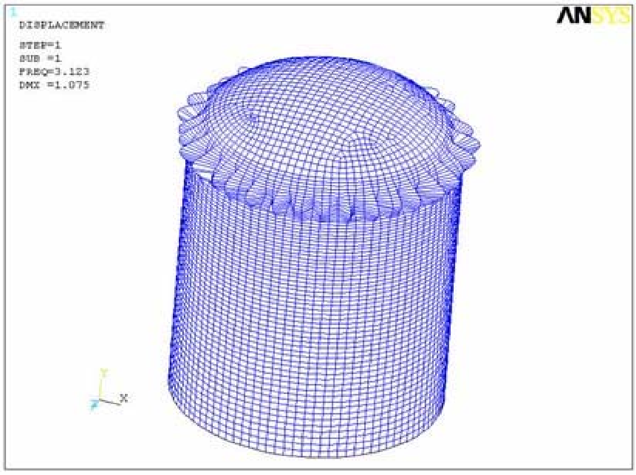


Fig. 2.2.1. Sketch of Steel Containment Vessel



(a) Hoop Stress of Axisymmetric Model



(b) Local Buckling of Steel Containment Model

Fig. 2.2.2. Hoop Stress and Buckling Mode of Steel Containment Loaded by Internal Pressure

Steel containment vessel is assembled with cylindrical shells and two semiellipsoidal heads (Fig 2.2.1.). When LOCA or serious accident happens and steam pressure in the containment increases, it is necessary that the steel containment be able to bear the pressure load and that steam pressure be confined within its allowable limit. Two aspects should be considered so as to obtain the ultimate pressure load-carrying capability of steel containment: one is to ensure that the stress or strain be controlled within its limit, the other is to ensure that the buckling load and displacements be limited within its allowable values.

Fig. 2.2.2. shows that compression stress will occur in some local regions of the steel containment when it is loaded by internal pressure, which causes local buckling. So coupling analysis taking account of both plasticity and buckling effects should be implemented to obtain the ultimate pressure load-carrying capability of the structure. In the design of AP1000, model test for pressure load-carrying capability of steel containment was implemented, and the load-carrying capability was also evaluated with analytical methods based on deterministic and probabilistic theory, which shows the bearing capability obtained from different methods accords with each other by and large[3].

2.3. In-Vessel Retention (IVR) of Molten Core Debris

When core meltdown accident happens, the molten core debris may be designed retaining in reactor vessel while the bottom of reactor vessel shouldn't get melted, which reduces the probability that radioactivity is let out so that accident sequence is mitigated. However, in order to retain molten core debris, whose temperature is up to one thousand, in the reactor vessel, cooling water outside the reactor vessel must be sufficient so as to cool the fervent lower head, and the steam that is heated should be released in time. Thus, the following two mechanical issues should be taken into account:

(1) Cooling and releasing issue of the core cavity water——it is a thermo-hydraulic problem concerning coupling between two-phase fluid and solid, that is, the problem about heat transfer and heat conduction between two-phase fluid injected and RPV solid.
(2) Creep failure of RPV at high temperature—— at high temperature, material durability limit will decrease as time passes and creep strain will increase. Material nonlinearity at high temperature and creep failure analysis are also difficulty of this item.

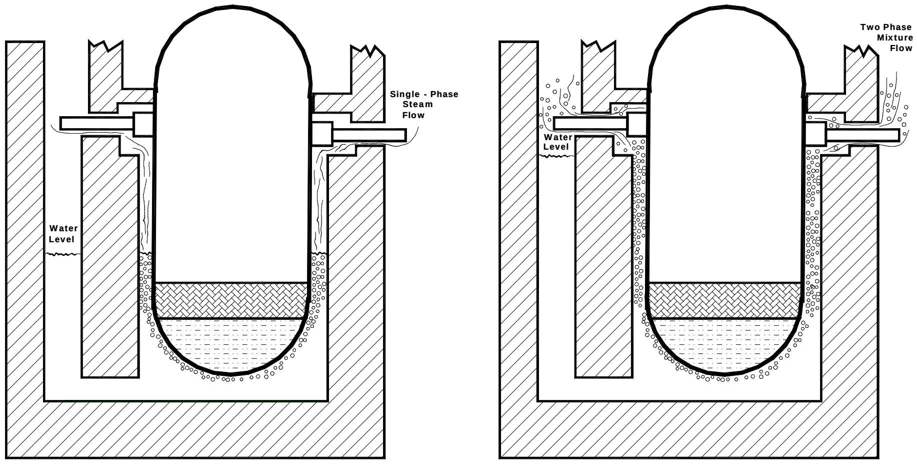


Fig. 2.3.1. IVR Issue

2.4. Seismic Design without OBE for NPP

One of the standardized designs of advanced PWR is seismic design standardization. To simplify seismic design, URD abrogates seismic design method of OBE, which reduces the total investment of seismic design from original 8~10% to below 5%.

The seismic design contents basically contain seismic classification in nuclear power plant, seismic requirements for OBE abrogation, seismic margin evaluation and seismic risk evaluation, etc.
These contents are described in reference [4] in detail, so they are not to be repeated here.

2.5. Thermo-Hydraulic Issues Concerning Coupling between Two-Phase Fluid and Solid

The replacement and supplement of passive safety systems ingeniously take advantage of the simple physical law of nature, "gravity", to design cooling in natural circulation form, which greatly reduces the quantity of facility and components that require energy raw material (electricity, oil, etc.) input. The replacement and supplement of passive safety systems also reduce failure probability of systems and equipments, make human control relatively simple, and increase general safety and economy.

However, in order to ensure that the passive system can maintain its natural circulation function, strict analyses and tests shall be scientifically implemented to solve thermo-hydraulic problems that concern coupling between two-phase fluid and solid. The problem of steel containment, cooled by cool water outside and circulated naturally with steam inside, is a typical example. The fluids are two-phased with gas and liquid states, and they are coupled with the containment shell to transfer heat, which makes it rather difficult to analyze.

Besides, the thermo-hydraulic fluid-structure coupling in IVR is more difficult to solve, so both mechanical and thermo-hydraulic engineers should put more emphasis on it.

3. MAIN MECHANICS FOR THE DESIGN OF GEN-IV SCWR

3.1. Main Features of Design

In May 2000, the Generation IV International Forum (GIF) selected 6 most promising reactor types as Gen-IV advanced reactor systems. Here we'll take example for "Supercritical Water-cooled Reactor" (SCWR), whose design principle is shown in Fig. 3.1.1. The pressure vessel and core construction form of SCWR are similar to that PWR, yet the coolant of SCWR works above the thermodynamic critical point of water (374℃, 22.1MPa) and this kind of "cooling water" has dual nature of liquid and gas, which makes the heat conduction efficiency better than that of ordinary light water. Compared with currently light water reactor, the heat efficiency of SCWR is enhanced by one third. The main reference values of SCWR parameters are listed in Table 3.1.1.[5].

3.2. Mechanical Difficulties Requiring Further Research

(1) The core inlet and outlet temperatures are 280° and 510℃, respectively, so the temperature increase of coolant inside the core (4~5 meters in height) is 230℃, which means the temperature increases rapidly. What's more, the coolant inside the core has to get through the critical point of 374℃, under which temperature physical properties of "water", such as its density, specific heat capacity, etc. will change suddenly (Fig. 3.2.1.(a)). Fluid dynamic response induced by the sudden change of thermo-hydraulic parameters near critical point will cause some severe influences listed as follow:

Table 3.1.1. Reference Value of SCWR Main Parameter

Parameter	Unit	Value
Power	MWe	1700
Coolant pressure	MPa	25
Coolant inlet temperature	℃	280
Coolant outlet temperature	℃	510
Net efficiency	%	44
Fuel		UO_2
Burnup	GWP/MTHM	45
Damage	dpa	10-30

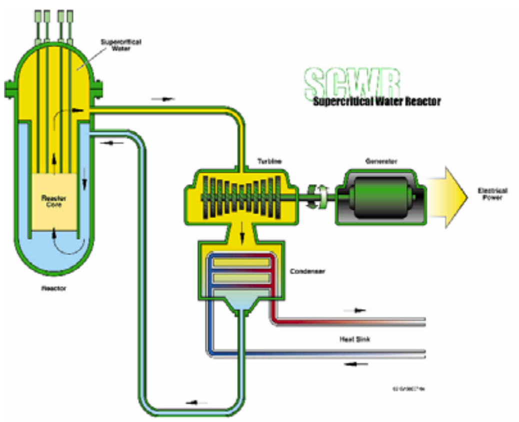
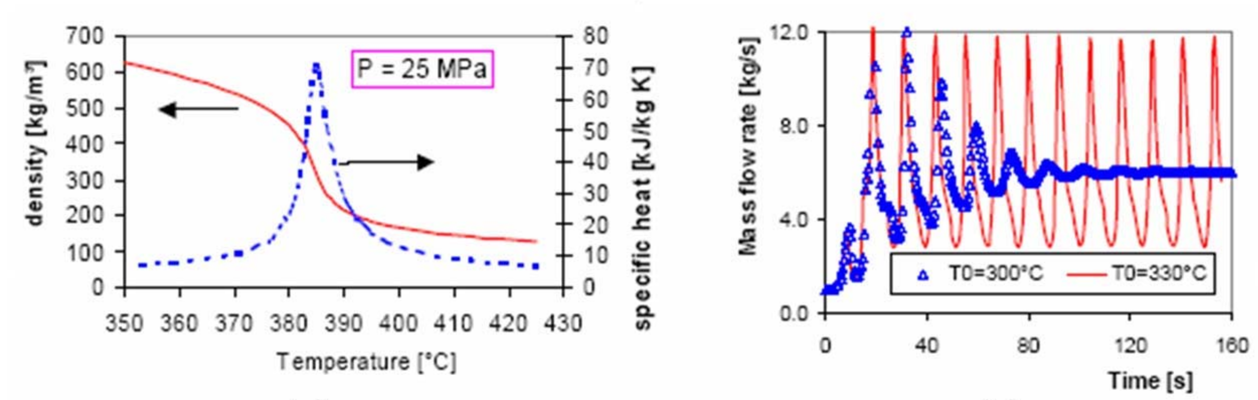


Fig. 3.1.1. Design Principles of SCWR 图

① "Shock wave"effect will occur near critical point, which means parameters such as pressure, density, etc., will change suddenly.

② Mass flow, pressure will fluctuate (Fig. 3.2-1(b)).

③ Structure “flutter” may be induced, which is a hundred times as severe as, if not more severe than, flow-induced vibration in current PWR.



（a）　　（b）

Fig. 3.2.1. Abnormal Change of Main Physical Parameter of Supercritical Water near Critical Point

Then, why don’t these problems occur in current commercial subcritical and supercritical fossil-fired power plants? We’ll take example for 600MWe supercritical fossil-fired power plant. The inlet and outlet temperatures are 280° and 541℃, respectively, and there’s no enclosed type pressure vessel in heating devices. The height range of heating space is 30~40 meters, and the distance that coolant flows along rifled heater is up to 500 meters (Fig. 3.2.2.), which indicates that the coolant is heated gradually and that sudden change of parameters mentioned above will be avoided at the critical point.

(2) The design temperatures of Gen-II or Gen-III reactor vessel are both below creep temperature of the material and influence of creep is not required to consider. However, the temperature and

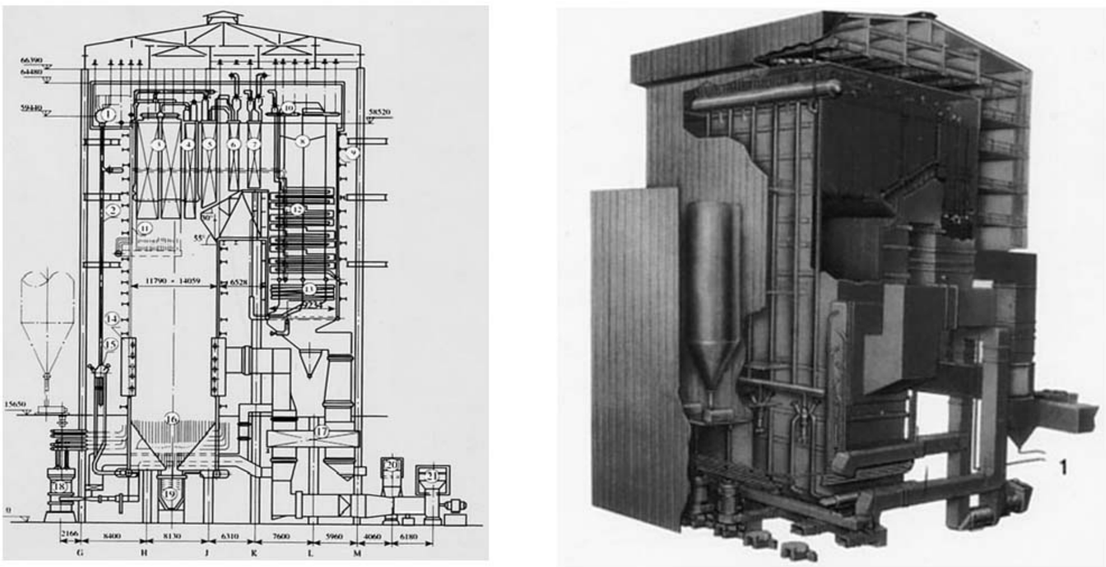


Fig. 3.2.2. Heating Sketch of Supercritical Fossil-Fired Plant

pressure of SCWR are even higher, so it is necessary to take account of the cumulative coupling damage effect associated with high-temperature creep and fatigue upon component life for coolant circle. For material of RPV core section, further consideration should be given to acceleration effect of creep induced by irradiation.

(3) Since no steam generators are designed in SCWR, supercritical water directly enters the steam turbine and does work. Hence the pressure boundary of steam turbine belongs to nuclear Class 1, which makes dynamic sealing of its rotor bearing a technical problem of operation and maintenance.

4. CONCLUSION

From aforementioned discussion on mechanical issues and technical difficulties associated with Gen-III passive NPP and Gen-IV advanced SCWR, we may clearly make out that the differences between mechanics of Gen-III or Gen-IV NPP and those of current NPP are as follows:

(1) Thermo-hydraulic dynamics and structural dynamics coupling problems become dominant.

(2) Creep at high temperature and fatigue coupling damage accumulation should be given attention to.

To solve these problems, we should utilize principles and methods of modern applied mechanics, i.e. traditional solid and structure mechanics, fluid mechanics, thermodynamics, material science, computer simulation technology, etc. and cross-disciplines of these subjects.

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