U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

US Presentation on AP1000 Review Plans and Status

MDEP Meeting

March 22-24, 2010

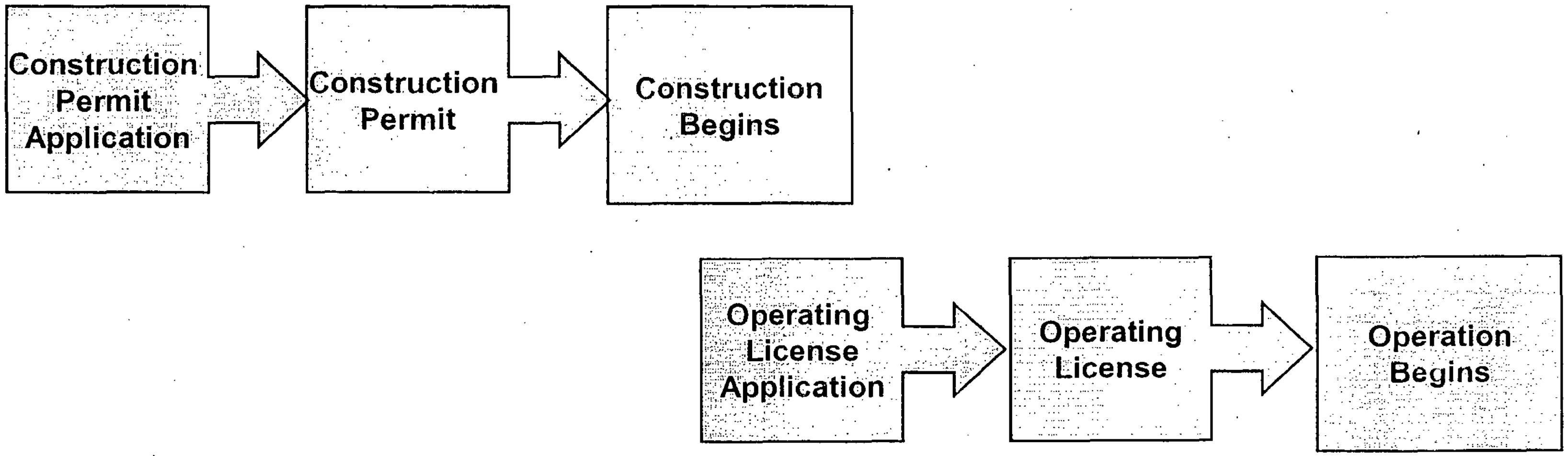
I/25

(7)

Overview

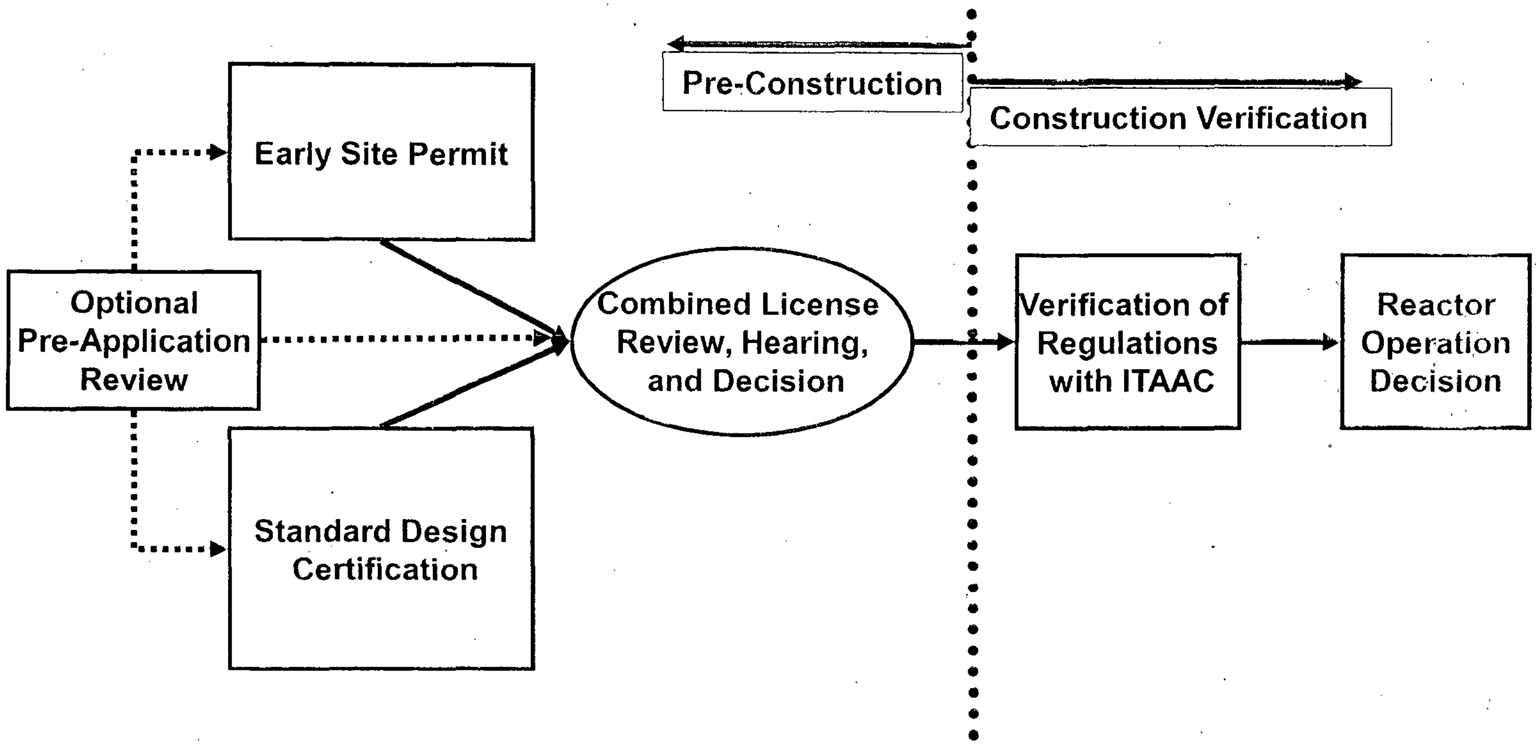
- Part 52 Licensing processes
- Design certifications and combined licenses
- Amendment review
- Status

Part 50 Licensing Process



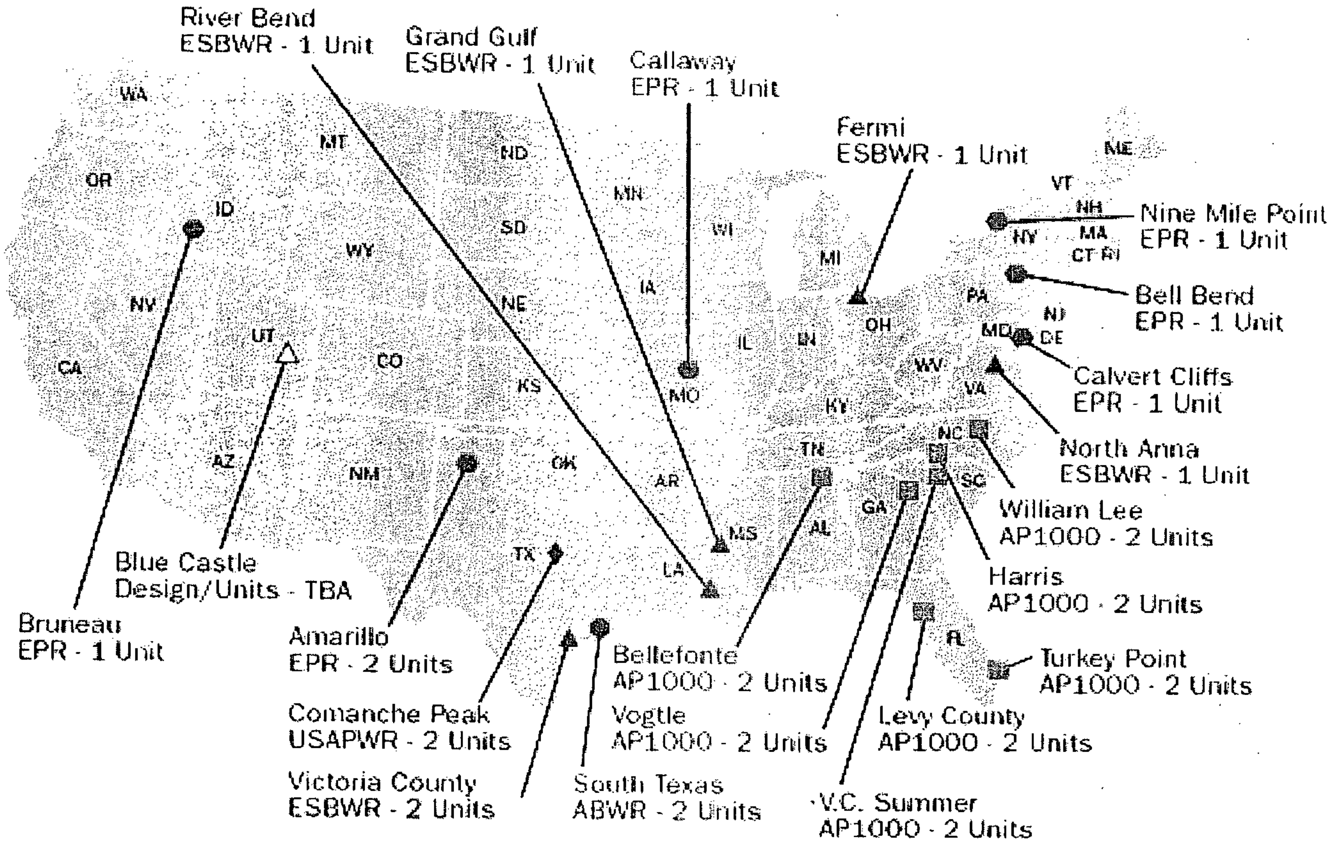
- Design effort proceeded throughout process
- No backfit protection with a Construction Permit
- Regulatory standards evolved as construction proceeded

Part 52 Objectives



- Licensing decisions finalized before major construction begins
- Inspections w/ITAAC to verify construction
- ~~Limited work may be authorized before COL issuance~~

New Reactor Applicants



You may click on a design name to view the NRC's Web site for the specific design.

● ABWR ▩ AP1000 ● EPR ▲ ESBWR ♦ USAPWR △ Design/Units - TBA

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Combined License Applications

- Combined construction permit and operating license for a nuclear power plant
- May reference an early site permit, a standard design certification, both, or neither
- Objective is to resolve all safety & environmental issues before authorizing construction
- Prior to fuel load, must verify the facility has been constructed in accordance with the license (ITAAC)

Amendment Review

- AP1000 design (Rev. 15) was certified by the NRC in Appendix D to part 52 in 2006.
- Westinghouse requested changes to the design (Rev. 16 in 2007, with rev. 17 in 2008)
- Design changes and COL action item responses
- NRC focus is on the changes, not a complete re-evaluation
- Supplement to previous SER will address the changes

Status

- Preparation of SE with open items complete for almost all chapters.
- Chapter 6 expected to be completed in June. Parts of chapter 3 related to seismic review also not complete. Shield building review is the critical path for schedule.
- Resolution of open items continues for most chapters.
- Some new design changes recently identified by Westinghouse.
- Revised schedule being developed

Major review areas

- Seismic analyses including analyses for soil sites and shield building modifications
  - Revised shield building submittals this spring
- Containment sump performance and downstream effects – review almost complete
- Control room ventilation and dose - complete
- Completion of human factors engineering, instrumentation and control and piping analysis underway, but behind schedule

Background of the CRDM Topic

by SUN Zaozhan

Background of the CRDM Topic

There are such kind of queries in the first batch of questions during the safety review of Sanmen NPP Unit 1&2's PSAR:

- What is the limit value of CRDM drop-time ? (the time after drive rod assembly starts to drop from the highest position of the stroke and before control rod assembly enters the dashpot section of control-rod guided tube of fuel assembly)
- What are the actual measurement results under the test conditions of hot/cold and earthquake?

Background of the CRDM Topic

The Applicant, of Construction Permit, answers that the AP1000 safety analyses utilize a rod drop time of 2.4 seconds. "Based on experience actual rod drop times are expected to be less than the value used in the safety analyses, however there have been no rod drop testing performed specifically for AP1000."

Background of the CRDM Topic

The Reviewer further asked in an Action Sheet, "What is the limited value of the drop-time of control rod assembly under the earthquake conditions?" The Reviewer also required the submission of CRDM's anti-seismic test report, but the Applicant did not give a straightforward answer.

Background of the CRDM Topic

According to HAF.J0053, a technical document by National Nuclear Safety Administration (NNSA), seismic qualification must be done by testing when analysis can not sufficiently prove the integrity and operability of seismic category I equipment, such as Control Rod Drive System (CRDS).

Background of the CRDM Topic

Since PSAR Sections 3.9.2.2 and 3.9.4 do not contain relevant information, and the Applicant did not give satisfactory explanations about it, the Reviewer repeated in a question of the second batch the requirement for describing the seismic qualification testing of CRDS.

Background of the CRDM Topic

The Applicant considers that pressure housing is the only Seismic Category I safety-related part of the CRDS. "As is stated in the first paragraph of 3.9.4.4 of the PSAR, these components will use analysis to prove the integrity and operability over the 60 year design life. No seismic qualification testing of the CRDS is required."

Background of the CRDM Topic

After several rounds of dialogues, the Applicant claimed that CRDM belongs to AP1000 Class D and non-seismic Category according to AP1000 safety classification system, and therefore anti-seismic tests are not required. The Applicant also stated that no drop time limits are presented to the CRDS during safe shutdown earthquakes.

Background of the CRDM Topic

Because that the CRDS performs the fundamental safety function, "to shut down the reactor", the Reviewer considers that it is not adequate to put CRDS into AP1000 Class D and non-seismic Category. Furthermore, the present practices in China adopt that CRDS belongs to safety-related items, with reactor coolant pressure boundary being Safety Class 1.

Background of the CRDM Topic

So, "CRDS should be classified as safety-related and seismic Category I item". But, in CRDS of AP1000, only the reactor coolant pressure boundary is Safety Class 1, with all other items being non-safety related and non-seismic Category. This is not consistent with the descriptions about safety related items in PSAR and the requirements in ANSI 51.1.

Background of the CRDM Topic

According to HAF.J0066, "Technical Positions on Classification of PWR SSCs", the names and symbols of the classes are as follows:

Class Name	Class Symbol
Safety Class	SC *(1E for electrical equipment)*
Safety Class 1	SC-1
Safety Class 2	SC-2
Safety Class 3	SC-3
Non-Safety Class	**NNS** *(Non 1E Class for electrical equipment)*
Non-Safety Class with special requirements	**NNS(S)** *(SR for electrical equipment)*

Background of the CRDM Topic

"Safety Class" in the above table applies to non-pressure retaining mechanical components, electrical equipments and buildings or civil structures.

CRDM is clearly assigned the "Safety Class".

Background of the CRDM Topic

It can be seen from the final safety analysis report (FSAR) that the safety classifications of Qinshan Nuclear Power Plant (QNPP) structures, systems and components are according to ANSI/ANS51.1—1983.

Background of the CRDM Topic

Table 3.2-1 in the FSAR shows that both the RCCAs and the CRDMs are classified Safety Class 3 and Seismic Category I, and the housings of the CRDMs are classified Safety Class 1 and Seismic Category I.

Background of the CRDM Topic

Many of the NPPs built in China, like Da Ya Bay, Ling Ao and so on, are designed according to French RCC standard series. From Table T-3.2-1 in the FSAR of Da Ya Bay NPP, it can be seen that the CRDMs are classified Safety Class LS and Seismic Category I, and the housings of the CRDMs are classified Safety Class 1 and Seismic Category I.

Background of the CRDM Topic

Tianwan Nuclear Power Plant (TNPP) is imported from Russia. The TNPP FSAR Subsection 3.9.4.1.1 says, the CPS drive system is a constituent part of the reactor control and protection system and could include up to 121 control rod drive mechanisms required for start-up, power control, excessive reactivity compensation and reactor shutdown by control rod insertion to or withdrawal out of the reactor core.

Background of the CRDM Topic

The CPS drive boundary is an interface of the CPS drive and the control rods. The CPS drive system together with the control rods present one independent system of reliable reactivity control under all the reactor plant operating conditions. "The CPS drive ШЭМ-3 has classification designation 2Н3У in accordance with /1/, refers to group B as per /2/ and seismic category I as per /3/. The drive is presented in figures F-3.9.4-1 and F-3.9.4-2."

Background of the CRDM Topic

The three referenced documents are:

- 1 Общие положения обеспечения безопасности атомных электростанций ОПБ-88/97, ПНАЭ Г-01-011-97, Москва, 1997. （核动力厂安全保障总则）
- 2 Правила устройства и безопасной эксплуатации оборудования и трубопроводов атомных энергетических установок ПНАЭ Г-7-008-89, Москва, 2000. （核动力装置的设备、管道的设置及安全运行规范）
- 3 Нормы проектирования сейсмостойких атомных станций ПНАЭ Г-5-006-87, Москва, Энергоатомиздат, 1989. （核动力厂抗震设计规范）

Background of the CRDM Topic

Section 4.2 of ANSI/ANS-51.1-1983 is about reactivity control systems. Subsection 4.2.1 specifies the nuclear safety function of these systems is to achieve and maintain the reactor core subcritical for any mode of normal operation or event or to introduce negative reactivity or limit the introduction of positive reactivity for any PC-3,-4, or -5 event.

Background of the CRDM Topic

Section 4.2 of ANSI/ANS-51.1-1983 is about reactivity control systems. Subsection 4.2.1 specifies the nuclear safety function of these systems is to achieve and maintain the reactor core subcritical for any mode of normal operation or event or to introduce negative reactivity or limit the introduction of positive reactivity for any PC-3,-4, or -5 event.

Subsection 4.2.2 defines "control rod drives, and those portions of the control rod drive units utilized for rapid insertion of the control rods" as components of reactivity control systems.

Background of the CRDM Topic

Subsection 4.2.4 specifies that the reactivity control systems pressure boundary equipment and supports, within the scope of ASME B&PVC, Section III and that provide the nuclear safety function of 4.2.1, shall be SC-2. Other equipment that provides the nuclear safety function of 4.2.1, shall be SC-3.

Background of the CRDM Topic

In Table A-1, “Equipment Classification”, of Appendix A, “Classification Application”, of ANSI/ANS-51.1-1983, “control element assemblies” and “control element drive mechanism” are assigned safety class of SC-3 and seismic requirement of SSE.

Background of the CRDM Topic

After almost a year of safety review, the Construction Permit was issued to Sanmen Nuclear Power Company (SMNPC) with the following condition: SMNPC complete the reasonability justification of CRDM's safety classification and seismic categorization , and explain how the assigned safety functions are satisfied, in half a year.

Thanks

Introductive description of the

PWR CRDM components

and the Understanding of their safety functions

by SUN Zaozhan

Introductive description of the

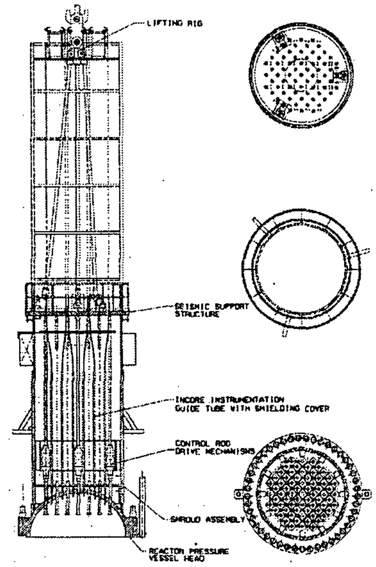
PWR CRDM components

and the Understanding of their safety functions

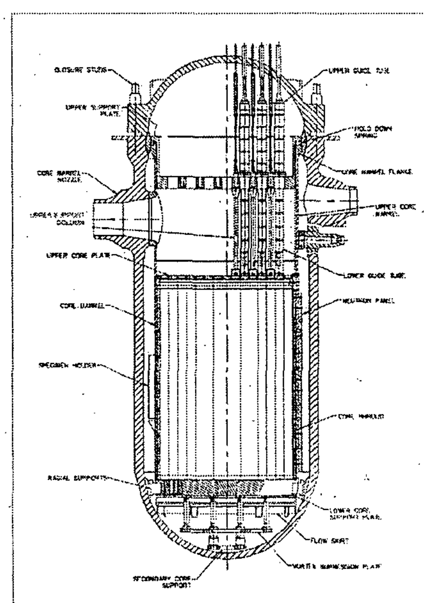
The AP1000 control rod drive mechanism (CRDM), is based on a proven Westinghouse design that has been used in many operating nuclear power plants.

To my knowledge, all the PWR NPPs use very similar CRDMs.

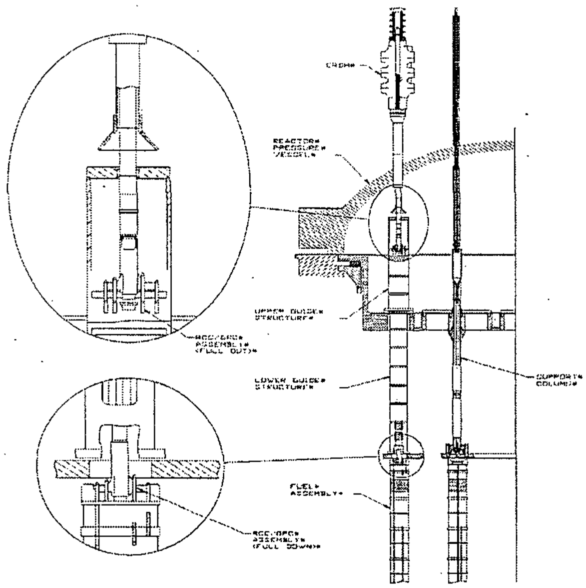
The PWR NPP's CRDMs are usually located on the head of the reactor vessel.



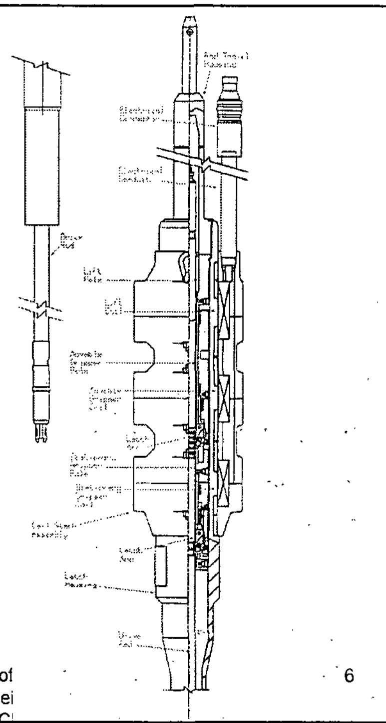
The CRDMs are coupled to rod cluster control assemblies (RCCAs) that have neutron absorber material over the active length of the control rods.

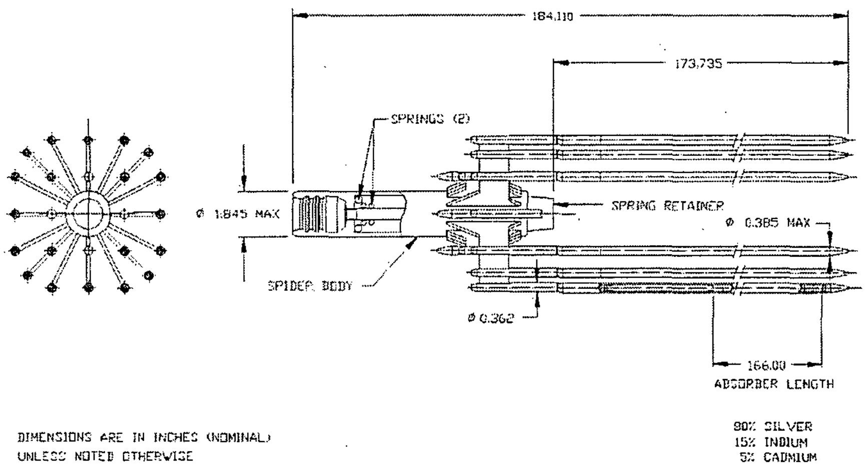


A more detailed description of the coupling of CRDMs with RCCAs and GRCAs.



A more detailed description of the CRDMs





A more detailed description of the RCCAs

Introductive description of the

PWR CRDM components and the Understanding of their safety functions

For AP1000, the primary functions of the CRDM is to insert or withdraw, at a designated speed, 53 rod cluster control assemblies and 16 gray rod control assemblies from the core to control average core temperature. During startup and shutdown the control assemblies control changes in reactivity.

Introductive description of the

PWR CRDM components and the Understanding of their safety functions

Operation of the control rod drive mechanisms is integrated to move groups of assemblies together. Each cluster assembly is in a bank of assemblies which is used for reactivity control, axial power distribution control, or shutdown control. The assemblies of each bank of several rod cluster control assemblies or gray rod control assemblies move at the same time.

Introductive description of the

PWR CRDM components and the Understanding of their safety functions

The design of the control rod drive mechanisms also permits holding the rod cluster control assemblies and the gray rod control assemblies at any step elevation within the range of rod travel during normal operation. The rod cluster control assemblies and gray rod control assemblies have the same mechanical coupling with the control rod drive mechanism.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

The control rod drive mechanism is a magnetically operated jack (magjack). A magnetic jack is an arrangement of three electromagnets energized in a controlled sequence by a power cycle to insert or withdraw rod cluster control assemblies and gray rod control assemblies in the reactor core in discrete steps.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

The CRDM is designed to release the drive rod and RCCA during any part of the power cycle sequencing if electrical power to the coils is interrupted. When released from the CRDM, the drive rod and rod cluster control assembly or gray rod control assembly falls by gravity into a fully inserted position.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

The control rod drive mechanism consists of four separate subassemblies. They are:

the pressure vessel,

the coil stack assembly,

the latch assembly, and

the drive rod assembly.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

The pressure vessel includes:

a latch housing and a rod travel housing that are connected by a threaded, seal-welded, maintenance joint that facilitates removal of the latch assembly.

The coil stack assembly includes:

the coil housings, electrical conduit and connector, and three operating coils: the stationary gripper coil, the movable gripper coil, and the lift coil.

The coil stack assembly is installed on the drive mechanism by sliding it over the outside of the latch housing.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

The latch assembly includes the guide tube, stationary pole pieces, movable pole pieces, and two sets of latches: the movable gripper latches and the stationary gripper latches. The latches engage grooves in the drive rod assembly.

Introductive description of the
PWR CRDM components and the Understanding of their safety functions

The movable gripper latches are moved up or down by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next step.

Introductive description of the
PWR CRDM components and the Understanding of their safety functions

The drive rod assembly includes a coupling, drive rod, disconnect button, disconnect rod, and locking button. The drive rod has a 5/8-inch (15.9 mm) pitch from groove to groove that engage the latches during holding or moving of the drive rod.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

The control rod drive mechanism withdraws and inserts a rod cluster control assembly or gray rod control assembly as shaped electrical pulses are received by the operating coils. Withdrawal of the drive rod and RCCA or GRCA is accomplished by magnetic forces. Insertion is by gravity.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

Preliminary Safety Analysis Report (PSAR) Section 3.2 of Sanmen NPP Unit 1&2, which are of AP1000 type reactor, gives the SSC's safety classification definitions. The following are written in the PSAR:

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

"**Safety-related** is a classification applied to items relied upon to remain functional during or following a design basis event to provide a safety-related function. Safety-related also applies to documentation and services affecting a safety-related item.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

- **Safety-related function** is a function that is relied upon during or following a design basis event to provide for the following:
- The integrity of the reactor coolant pressure boundary
- The capability to shut down the reactor and maintain it in a safe shutdown condition
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

Design basis event is an event that is a condition of normal operation (including anticipated operational occurrences), a design basis accident, an external event, or natural phenomena for which the plant must be designed so that the safety-related functions are achievable."

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

To my understanding, some of the CRDMs are relied upon during or following a DBE to provide the capability to shut down the reactor and maintain it in a safe shutdown condition, or they are relied upon to remain functional during or following a DBE to provide a safety-related function. Therefore, these CRDMs should be classified as safety-related items.

Introductive description of the

PWR CRDM components
and the Understanding of their safety functions

Besides, Section 3.2 of the PSAR also gives the following descriptions:

"Class B is a safety-related class equivalent to ANS Safety Class 2. This class is designed to accomplish the following:

Introductive description of the

PWR CRDM components
and the Understanding of their safety functions

- ....
- It introduces emergency negative reactivity to make the reactor subcritical (for example, control rods).
- ....

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

Class C is a safety-related class equivalent to ANS Safety Class 3. Class C applies to structures, systems, and components not included in Class A or Class B that are designed and relied upon to accomplish one or more of the following safety-related functions:

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

- ....
- Provide for functions defined in Class B where structures, systems, and components, or portions thereof are not within the scope of the ASME Code, Section III, Class 2.
- ...."

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

Keep all these in mind:

CRDM provide for such a function as to "introduces emergency negative reactivity to make the reactor subcritical", which is defined in Class B

What makes us think so is that the control rods are more or less relying on the CRDMs to introduce emergency negative reactivity.

Introductive description of the
PWR CRDM components
and the Understanding of their safety functions

But, CRDMs are components not within the scope of the ASME Code, Section III, Class 2.

So, they should be classified as Class C.

Thanks

Practices and Regulatory Requirements on CRDM Qualifications in China

by SUN Zaozhan

Practices and Regulatory Requirements on CRDM Qualifications in China

The Chinese regulatory document HAF102, Code on the Safety of Nuclear Plant Design, which is originated from the IAEA document NS-R-1, Safety of Nuclear Power Plants: Design, stipulate in Section 5.5, Equipment Qualification, that

Practices and Regulatory Requirements on CRDM Qualifications in China

"A qualification procedure shall be adopted to confirm that the items important to safety are capable of meeting, throughout their design operational lives, the demands for performing their functions while being subject to the environmental conditions (of vibration, temperature, pressure, jet impingement, electromagnetic interference, irradiation, humidity or any likely combination thereof) prevailing at the time of need.

Practices and Regulatory Requirements on CRDM Qualifications in China

The environmental conditions to be considered shall include the variations expected in normal operation, anticipated operational occurrences and design basis accidents. In the qualification programme, consideration shall be given to ageing effects caused by various environmental factors (such as vibration, irradiation and extreme temperature) over the expected lifetime of the equipment.

Practices and Regulatory Requirements on CRDM Qualifications in China

Where the equipment is subject to external natural events and is needed to perform a safety function in or following such an event, the qualification programme shall replicate as far as practicable the conditions imposed on the equipment by the natural phenomenon, either by test or by analysis or by a combination of both."

Practices and Regulatory Requirements on CRDM Qualifications in China

Equipment qualification comprises two parts:

seismic and environmental.

Only those components satisfy the qualification criteria can be installed in NPPs.

Practices and Regulatory Requirements on CRDM Qualifications in China

In the very first stage of China's civilian nuclear power utilization, the standards such as IEEE Std 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations", IEEE Std 382, "IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants", IEEE Std 344, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations", were used for equipment qualification.

Practices and Regulatory Requirements on CRDM Qualifications in China

China's first NPP was built in the early 1980s and NNSA was founded in 1984. Beginning in that time, NNSA requires all the safety-related components be qualified before installation on the site.

Practices and Regulatory Requirements on CRDM Qualifications in China

Based on experiences of seismic qualification of active mechanical components, such as pumps and valves, and Class 1E electrical equipments, NNSA issued in 1995 the technical document HAF-J0053, Guidance on Seismic Qualification Tests for Nuclear Class Equipments.

Practices and Regulatory Requirements on CRDM Qualifications in China

From then on, the seismic qualification tests stepped on a relatively normalised road. In 1997, a national code GB50267-97, Code for Seismic Design of Nuclear Power Plants, was issued. From then on, China has its own normative document for seismic analysis of nuclear components.

Practices and Regulatory Requirements on CRDM Qualifications in China

- For environmental qualification, the standards used remained to be IEEE's, RCC's, or KTA's.

- The equipment qualification consists of the following steps:

Practices and Regulatory Requirements on CRDM Qualifications in China

First, a list of equipment or components, including active mechanical components, Class 1E electrical equipment and non-metallic appurtenances (seals, fillings, paintings, lubricants), need to be qualified should be established according to their safety classifications.

Practices and Regulatory Requirements on CRDM Qualifications in China

The list should reflect types, loads (pressure, temperature, mechanical load, radiation, et cetera), manufacturers, and locations in the plant buildings. If a component belongs to reactor coolant pressure boundary, it should be specifically noted.

Practices and Regulatory Requirements on CRDM Qualifications in China

Second, prototypes representing different kinds of equipment or components from the list should be selected. The factors considered for the selection include equipment type, anticipated loadings, size, and manufacturer.

Third, the qualifications should finally be implemented according to adequate codes or standards.

Practices and Regulatory Requirements on CRDM Qualifications in China

The qualification types in China are classified as follows:

For electrical equipment, there are three types of qualifications, with their names being K1, K2 and K3 respectively.

Practices and Regulatory Requirements on CRDM Qualifications in China

K1 qualification applies to those installed inside the containment and deemed to be functional during and after normal operation, anticipated operational occurrences, design basis accidents and safe shutdown earthquakes.

Practices and Regulatory Requirements on CRDM Qualifications in China

K2 qualification applies to those installed inside the containment and deemed to be functional during and after normal operation, anticipated operational occurrences and safe shutdown earthquakes, but not required to functional during and after design basis accidents.

Practices and Regulatory Requirements on CRDM Qualifications in China

K3 qualification applies to those installed outside the containment and deemed to be functional during and after normal operation, anticipated operational occurrences and safe shutdown earthquakes.

Practices and Regulatory Requirements on CRDM Qualifications in China

For mechanical components, there are two types of qualifications, with their names being M1, and M2. M1 qualification applies to those installed inside the containment and deemed to survive the design basis accidents and safe shutdown earthquakes.

Practices and Regulatory Requirements on CRDM Qualifications in China

Further distinctions among this type could be "operable" or "non-operable" during and after design basis accidents. M2 qualification applies to those installed outside the containment and deemed to survive the design basis accidents and safe shutdown earthquakes.

Practices and Regulatory Requirements on CRDM Qualifications in China

Although the qualification method can be analysis, test or experience, the environmental qualifications are, as a general rule, by test. For the seismic tests, analysis approach can be used for those equipment or components only have integrity requirements but no operability requirements, or for those having too big sizes to be seismic tested.

Practices and Regulatory Requirements on CRDM Qualifications in China

Seismic qualification must be done by tests when analyses can not sufficiently prove the integrity and operability of seismic category I equipment or components, such as Control Rod Drive System (CRDS), pumps, valves, relays, electrical equipment and measurement instrumentations.

Practices and Regulatory Requirements on CRDM Qualifications in China

As stipulated in IEEE Std 323, for most equipment, the following steps and sequence are considered acceptable:

- a) Inspection shall identify the test sample and ensure that it is not damaged.
- b) Specified baseline functional tests shall be performed under normal conditions.

Practices and Regulatory Requirements on CRDM Qualifications in China

- c) The test sample shall be operated to the extremes of all performance, operating, surge voltages, and electrical characteristics given in the equipment specifications, excluding design basis event and post-design basis event conditions, unless these data are available from other tests (e.g., design verification tests) on identical or similar equipment.

Practices and Regulatory Requirements on CRDM Qualifications in China

- d) When required, the test sample shall be age conditioned to simulate its functional capability at the end of its qualified life. Measurements made during, or baseline tests following, age conditioning can verify that the test sample is performing satisfactorily prior to subsequent testing. If condition monitoring is to be used in service, measurements after age conditioning would establish the qualified end condition.

Practices and Regulatory Requirements on CRDM Qualifications in China

- e) The test sample shall be subjected to specified nonseismic mechanical vibration.
- f) The test sample shall be subjected to simulated OBE and safe shutdown earthquake (SSE) seismic vibration in accordance with IEEE Std 344-1987.

Practices and Regulatory Requirements on CRDM Qualifications in China

- g) The test sample shall perform its required safety function(s) while exposed to simulated accident conditions, including conditions following the accident for the period of required equipment operability, as applicable. Accident radiation may have been included in step d). Safety function performance during testing shall be monitored. Note that safety function can be different in different stages of an accident.

Practices and Regulatory Requirements on CRDM Qualifications in China

- h) Post-test inspection shall be performed on the test sample, and all findings shall be recorded.

Because equipment or components are installed on floors, the seismic tests are generally implemented on shake tables, which generate simulated earthquake motions having response spectra enveloping the floor spectra.

Practices and Regulatory Requirements on CRDM Qualifications in China

But, for the CRDS, seismic tests with multi-excitations are considered more suitable. Such kind of tests was conducted in Nuclear Power Institute of China for CRDSs used in Qinshan Second Nuclear Power Plant and in China's Experimental Fast Reactor.

The CPS drive ШЭМ-3 was also seismic tested with multi-excitations.

Practices and Regulatory Requirements on CRDM Qualifications in China

In China's past experiences, the control rod drop time during safe shutdown earthquakes (SSEs) were required to be verified to meet a limit value, which is determined by safety analysis results against design basis accidents (DBAs).

Practices and Regulatory Requirements on CRDM Qualifications in China

But, during the safety review process of Sanmen NPP, the Applicant argued that DBAs and SSEs are not postulated to happen concurrently, and therefore the safety analyses against DBAs do not require a control rod drop time limit value. The reviewer accepted the logic in the argument, and requires that margins should be taken into account against earthquake conditions.

Practices and Regulatory Requirements on CRDM Qualifications in China

Before the seismic test, baseline functional tests and service life tests should be completed. These tests are usually also compared against the requirements in the USNRC document, Safety Review Plan (SRP). The following is included in Subsection 3.9.4 of SRP as for CRDS review guidance.

Practices and Regulatory Requirements on CRDM Qualifications in China

"Since the CRDS is a system important to safety and portions of the CRDS are a part of the reactor coolant pressure boundary (RCPB), General Design Criteria 1, 2, 14, and 29 and 10 CFR Part 50, § 50.55a, require that the system shall be designed, fabricated, and tested to quality standards commensurate with the safety functions to be performed, so as to assure an extremely high probability of accomplishing the safety functions either in the event of anticipated operational occurrences or in withstanding the effects of postulated accidents and natural phenomena such as earthquakes."

Practices and Regulatory Requirements on CRDM Qualifications in China

The control rod drive mechanisms of a new design or configuration should be subjected to a life cycle test program to determine the ability of the drives to function during and after normal operation, pressure testing, anticipated operational occurrences, seismic, and postulated accident conditions over the full range of temperatures, pressures, loadings, and misalignment expected in service.

Practices and Regulatory Requirements on CRDM Qualifications in China

The tests should include functional tests to determine times of rod insertion and withdrawal, latching operation, scram operation and time, system valve operation and scram accumulator leakage for hydraulic CRDS, ability to overcome a stuck rod condition, and wear. Rod travel and number of operational trips and test trips expected during the mechanism operational life should be duplicated in the tests."

Practices and Regulatory Requirements on CRDM Qualifications in China

For so long a period, the phrase "should be duplicated" in the above paragraph is considered as "should be doubled". The interpretation "should be copied" is considered as the real meaning of the phrase "should be duplicated" until the consultation to a NRC expert during a lecture course on AP1000 in China.

Practices and Regulatory Requirements on CRDM Qualifications in China

The design of EPR's CRDM is basically the same as that used in Germany, which has good operation experiences of more than 35 years. As for the anticipated life time, the Applicant of Taishan NPP explained that the CRDM line has successfully run 9×10^6 steps and more than 1000 drop tests were performed during KOPRA tests and so we can assure a service life of 6×10^6 steps.

Practices and Regulatory Requirements on CRDM Qualifications in China

It is said in PSAR Subsection 3.6.4.0.3.1.6 that "The release function of the Drive Rod (SCRAM) has to be ensured during and after earthquake. Furthermore the requirements for the drop time shall be ensured."

So, the Reviewer required the Applicant to explain how is ensured the above mentioned release function and the drop time requirements during and after earthquake.

Practices and Regulatory Requirements on CRDM Qualifications in China

The Applicant responded this way: "Due to the operating principle of the Latch Unit (sub-component of the CRDM maintaining the Drive Rod with coupled RCCA) the Drive Rod will be released and falls because of gravity into the core when the electrical power supplying the Latch Unit will be shut down. Additionally the releasing function of the Latch Unit is supported by springs. The release function of the Latch Unit is ensured as long as there are no plastic deformations of the Pressure Housing."

Practices and Regulatory Requirements on CRDM Qualifications in China

"The Pressure Housing belongs to the primary pressure boundary and contains the Latch Unit. It is shown in calculations that the deformations of the Pressure Housing stay in the elastic domain. In tests deformations were applied just to the Latch Unit simulating the deformations resulting from earthquakes. These tests showed no constrain to the release function of the Latch Unit with maximum calculated deformations."

Practices and Regulatory Requirements on CRDM Qualifications in China

"Other tests were also done deforming the Pressure Housing of the CRDM and checking the drop time. These tests proved the achievement of the drop time requirements during seismic excitations. By the different tests applying deformations it is shown that the release function and drop time is given during earthquake. By the proof that the deformations stay in the elastic domain it is shown that the requirements will be achieved after earthquake "

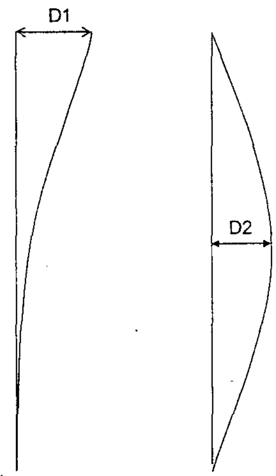
Practices and Regulatory Requirements on CRDM Qualifications in China

Because the above response explained the drop time requirement during earthquake by static tests taking into consideration of the calculated deformations, the Reviewer required to explain how to take into account the impact of the dynamic effect during earthquakes.

The Applicant then explained that the tests are in fact quasi-static.

Practices and Regulatory Requirements on CRDM Qualifications in China

Following tests were done:
Once the top of the CRDM was elastically deformed by a specified value of maximum D1. The other time the mid point of the CRDM was elastically deformed by a value of maximum D2. This lead in consequence to an elastic deformation of all the CRDM parts like during an earthquake.



Practices and Regulatory Requirements on CRDM Qualifications in China

During these tests the Drive Rod of the CRDM was located in the upper most position. After deflecting the top/mid point of the CRDM the device used for the deflection was removed. Now the top/mid point of the CRDM was able to make a free oscillating movement simulating the dynamic effect during an earthquake.

Practices and Regulatory Requirements on CRDM Qualifications in China

During this movement the Drive Rod was released and Drop time measured. Only small impacts on the Drop time were measured compared to an undeformed, not moving CRDM. Drop times showed good results and ability for drop was proven.

Thanks

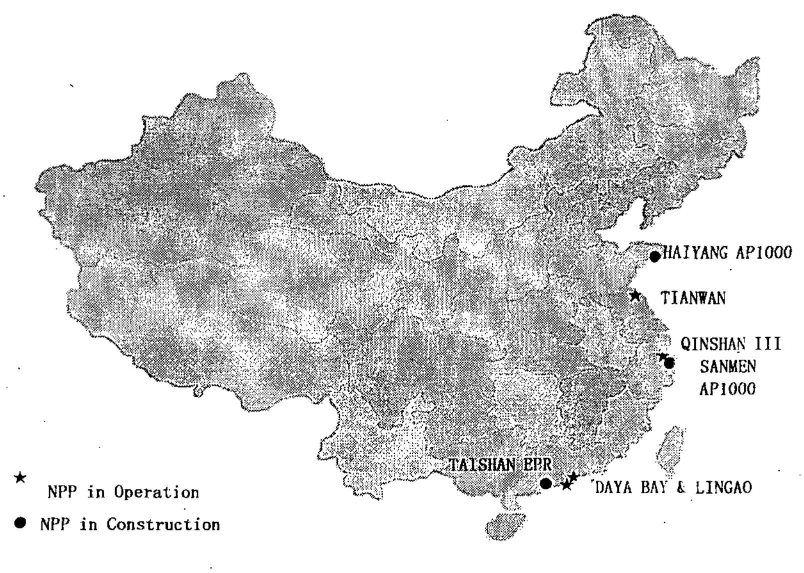
The Review of AP1000 in China

Chai Guohan

The Review of AP1000 in China

Background

- P.R.China and the U.S.A. signed a Frame Contract about import four AP1000 units on July 24, 2007 in Beijing, two units are assigned as Sanmen NPP unit 1&2, another two units are assigned as Haiyang NPP unit 1&2.
- The first engineering project of AP1000 in the world was started on December 31, 2007.
- Sanmen NPP submitted a CP application for its unit 1&2 to NNSA on February 25, 2008, with all supporting documents, including PSAR. Sanmen NPP expects that they can start construction FCD (First Pour of Concrete Delivery) in March, 2009, and connect to the grid in September, 2013.
- Haiyang NPP submitted a CP application for its unit 1&2 to NNSA on May 26, 2008, with all supporting documents, including PSAR. Hiaynag NPP expects that they can start construction FCD (First Pour of Concrete Delivery) in September, 2009, and connect to the grid in March, 2014.



licensing process of NPP in China

- licensing process of NPP in China is different from that of the US 10CFR52 (similar to 10CFR50 process).
- Two stages:
  - PSAR, Application for CP.
  - FSAR, Application for First fuel loading. After one year of trial operation, Application for OL.

Format and Content of PSAR

- PSAR of Sanmen NPP and Haiyang NPP are based on Tier 2 of AP1000 DCD Revision 16, with some plant specific information.
- The format and content of PSAR shall comply with RG1.70 and SRP.

Review Team

- 182 technical persons are involved, they are from Nuclear and Radiation Safety Center (NSC), Nuclear Power Research and Design Institute of China (NPIC), Beijing Nuclear Safety Review Center (subpart of Beijing Institute of Nuclear Engineering, BINE), Mechanical Equipment Reliability Center, Suzhou Nuclear Safety Center.
- NSC takes the overall technical responsibility of the nuclear safety review project, organizes and cooperates with other organization to carry out the review project.

Regulatory Basis (1/2)

- Due to some new concepts such as passive safety features are adopted in AP1000 design, the nuclear safety codes and standards which are effective now are not fully applicable. A lot of Policy Statements were published during review of AP600 and AP1000 in the US.
- After several months of Safety Review of AP1000, NNSA realized the important of this Issues. NNSA issued an Policy Statement (Technical Position) for the Review of First four AP1000 units in China.
- This Policy Statement was discussed during the Consultant Meeting of Nuclear Safety and Radiation Advisory Commission on March 6-7, 2009.

Regulatory Basis (2/2)

- According to the Policy Statement of NNSA, The Regulatory basis for Sanmen NPP unit 1&2 and Haiyang NPP unit 1&2 are:
  - Chinese Laws, Administrative Rules, forced National Standard should be followed. Nuclear Safety Requirements of NNSA should be met. Nuclear Safety Guides of NNSA shall be referenced.
  - US Codes, Rules applicable to AP1000 should be followed. Regarding the PSAR Contents which are identical to that of AP1000 DCD tier 2 Revision 15, US Codes, Regulatory Guides and Standards adopted by NRC in Review and Approval the AP1000 DC should be followed. The Policy Statement of NRC and American Practice should be followed. Regarding the design modification comparing to DCD revision 15, and COL Actions, latest and effective US Codes, Regulatory Guide and Standard applicable to AP1000 should be followed, latest NRC Policy Statement Applicable to AP1000 should be followed.

Safety Review Progress of AP1000 in China

	PSAR of Sanmen NPP			PSAR of Haiyang NPP	
	Initial Time Schedule	Adjusted Time Schedule	**Actual Progress**	Initial Time Schedule	**Actual Progress**
Receive CP Application with PSAR			**2008. 2. 27**		**2008. 5. 26**
Send Out Form Review RAI	2008. 3. 31		**2008. 3. 31**		**NA**
Receive Response to Form Review RAI			**2008. 5. 23**		**NA**
Send Out First Batch of RAI	2008. 5. 30		**2008. 5. 30**	2008. 10. 31	**2008. 10. 31**
Receive Response to First Batch of RAI	2008. 6. 30	2008. 7. 25	**2008. 7. 25**	2008. 12. 1	**2008. 11. 25**
First Review Meeting, Action Sheet	2008. 7. 15	2008. 8. 11	**2008. 8. 11–15**	2008. 12. 15	**2008. 12. 6–7**
Receive Response to Action Sheet of First Review Meeting	2008. 8. 15	2008. 9. 15	**2008. 9. 16**	2009. 1. 15	**2009. 1. 15**
Send Out Second Batch of RAI	2008. 9. 15	2008. 10. 15	**2008. 10. 17**	2009. 2. 16	**2009. 2. 20**
Receive Response to Second Batch of RAI	2008. 10. 15	2008. 11. 14	**2008. 11. 17**	2009. 3. 16	**2009. 3. 18**
Second Review Meeting, Action Sheet	2008. 10. 30	2008. 12. 1	**2008. 12. 3–5**	2009. 4. 1	**2009. 4. 7–8**
Receive Response to Action Sheet of Second Review Meeting	2008. 11. 30	2009. 1. 5	**2009. 1. 5**	2009. 5. 4	**2009. 5. 8**
Determine the Review Topics	2008. 12. 15	2009. 1. 20	**2009. 1. 23**	2009. 5. 20	**2009. 5. 20**
First Topical Review Meeting			**2009. 2. 3–4**		**2009. 6. 15**
Second Topical Review Meeting			**2009. 3. 9–10**		**2009. 7. 21**
Consultant Meeting of Nuclear Safety and Radiation Advisory Commission	2009. 1. 15	2009. 2. 16	**2009. 3. 13–14**	2009. 8. 5	**2009. 9. 23**
Issue CP	2009. 3. 18	2009. 3. 18	**2009. 3. 26**	2009. 9. 18	**2009. 9. 26**

Statistical Data of the Review

- Sanmen NPP
  - There are 1572 questions in the first batch of RAIs.
  - 663 Action Sheets are generated during the first review meeting.
  - There are 698 questions in the second batch of RAIs.
  - 269 Action Sheets are generated during the second review meeting.
  - 51 topical questions.
- Sanmen NPP
  - There are 856 questions in the first batch of RAIs.
  - 276 Action Sheets are generated during the first review meeting.
  - There are 239 questions in the second batch of RAIs.
  - 100 Action Sheets are generated during the second review meeting.
  - 52 topical questions.

Key Issues (1/2)

- Closed Issues
  - Version of the American Codes, Regulatory Guides, Standards adopted in the design.
  - Reliability of In-Vessel Retention Approach.
  - Impact of Commercial Aircraft to NI building.
  - Flow Induced Vibration test.
  - In Service Inspection of RCP flywheel.
  - Safety Margins of Containment Pressure in case of DBA.
  - Impact of dissolved Nitrogen to performance of PRHR.
  - Safety Classification of Equipments & Components of ECCS.
  - Selection of Accident Monitoring Parameters.
  - 3s Power Supply to RCP maintained after Turbine Trip.
  - Impact of SSE to control rod drop time after reactor scram

Key Issues (2/2)

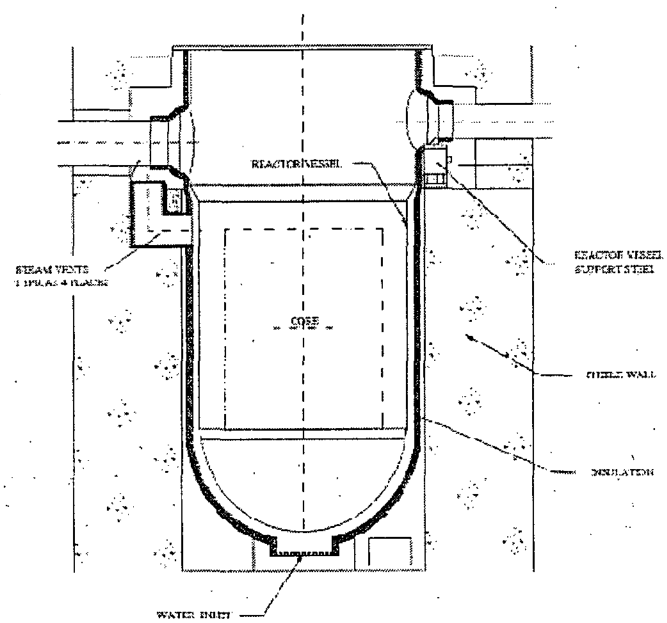
- Open Issues
  - COL Actions provided in Appendix F of NUREG-1793
  - Modification of Shield Building Structure
  - Strainer Design of AP1000 sump
  - The height of the exhaust vent stack
  - Specific ITAAC for Sanmen NPP

Version of US Codes, Regulatory Guides and Standards

- The version of some RGs, NUREGs and SRP listed in PSAR Section 1.9 is old, there are new Revision now, such as RG1.20r3, and new SRP.
- According to Chinese Nuclear Safety Practice, latest effective (valid) version should be adopted.
- WEC provide "AP1000 Assessment of Revised and New U.S. Nuclear Regulatory Guides" (APP-GW-GLR-800, Rev.0) and "AP1000 Conformance with SRP acceptance Criteria" (WCAP-15799).
- According to Policy Statement of NNSA, it is acceptable.

Reliability of IVR Approach

- IVR Approach is used for long term cooling of the Reactor Vessel. Two Issues which may block the flow should be considered:
  - Accumulation of Debris in the annulus space of IVR
  - Possible boron precipitation in the annulus space of IVR
- WEC reply
  - The total quantity of debris is small
  - No narrow area
  - Water Level in the cavity is enough
  - Two phase flow with high velocity in the annulus space
- We think it is acceptable.



Impact of Commercial Aircraft to NI building

- Issue concerned after 911 event , it is BDBA.
- Modification of Shield Building Structure in AP1000 DCD Revision 16.
- 10CFR50.150 (NRC-2007-0009) was published on June 12, 2009）. NRC don't have any comments about SC structure of shield building before that time.
- Our consideration:
  - It seems that SC structure is more strong than RC structure, but it lacks of standards and experience
  - We expect that NRC will accepted SC structure soon at that time.
  - The impact of Commercial Aircraft to NI building is not required to be considered now in China.

Flow Induced Vibration test

- Revision 2 of RG1.20 is adopted in PSAR, but the latest version of RG1.20 is Revision 3.
- "AP1000 Assessment of Revised and New U.S. Nuclear Regulatory Guides"（APP-GW-GLR-800，Rev.0）states:
  - Revision 3 does not modify the guidance in Revision 2, but does provide new guidance for steam dryers in boiling water reactors (BWRs) plants and information for monitoring programs for plant components outside the reactor vessel.
  - Should Revision 3 of Regulatory Guide 1.20 be implemented, there is a potentially significant impact on the design and licensing documentation for the AP1000 certified design. Westinghouse assessment is that design and testing complications of implementing Revision 3 guidance far outweigh any benefits.
- We think it is acceptable.

In Service Inspection of RCP flywheel

- For traditional PWR, ISI of RCP flywheel is considered.
- But it is not considered for AP1000 RCP.
- WEC explain:
  - Technical specification of material and ultrasonic examination during manufacture will provide the quality of flywheel.
  - Each flywheel is subject to a spin test at 125 percent overspeed, to demonstrate the quality of flywheel.
  - Pump shell can withstand the impact of flywheel , so it will not damage the integrity of RCS.
- We think it is acceptable.

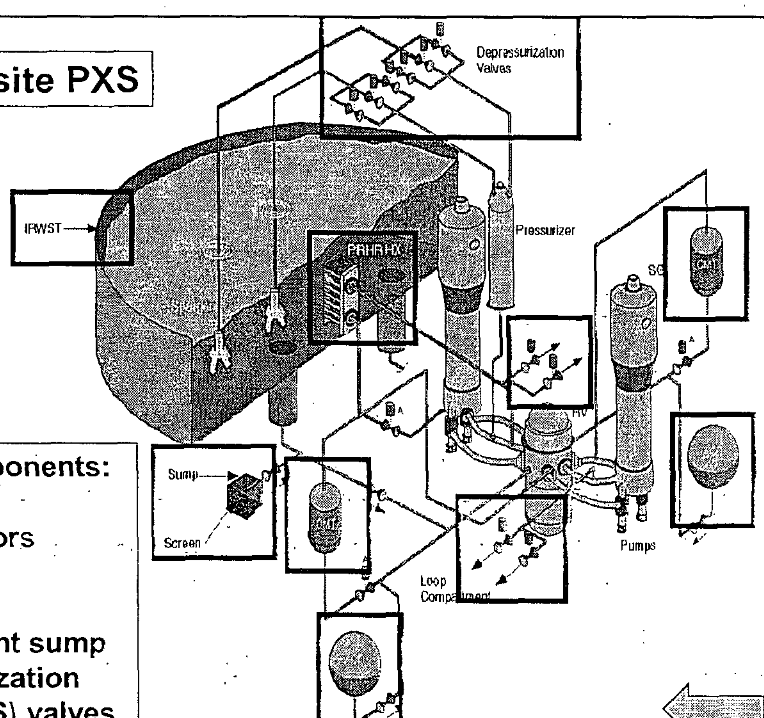
Safety Margin of Containment pressure in case of DBA

- The Design pressure of AP1000 containment is 0.407 MPa(g), but the analysis result shows the peak pressure during LB-LOCA is 0.399MPa, safety margin is 1.966%, can't meet the requirement of 10% at CP stage stated in SRP6.2.1.1A.
- NRC explain:
  - The requirement developed at a time when one supplier provided the NSSS and another supplier provided the containment, and at the CP stage the design was not complete. This is not the case for AP1000 where Westinghouse has designed both the NSSS and containment and the design of the standard certified plant is complete.
  - The containment pressure analysis of record was performed in a conservative manner and that more realistic analysis would show margin.
- We think it is acceptable.

Impact of dissolved nitrogen to the performance of PRHR

- Analysis result shows that Accumulator will put into operation during MSLB, nitrogen dissolve in water will release into the RCS, and it maybe accumulate in the PRHR, then it will impact the performance of PRHR.
- WEC reply:
  - The quantity of water injected from Accumulator is small.
  - Pressure difference between accumulator and RCS is small.
  - The SI water must pass through the majority of RCS prior to reaching the PRHR Hx.
  - There is a vertical tube at high point of PRHR to collect the non condensable gas.
  - The impact to the performance of PRHR is not significant.
- We think:
  - Risk of MSLB is very low.
  - If BE methodology adopted, the RCS pressure will not decrease to the degree of Accumulation injection.
  - Concept of defend-in-depth
- The risk is acceptable, so this issue can be closed.

Composite PXS



Major components:
CMTs
Accumulators
IRWST
PRHRHX
Containment sump
Depressurization (ADS) valves

Safety Classification of Equipment & Component of ECCS

- According to Chinese regulatory guide and US RG1.26, the safety classification of ECCS is safety class 2, quality group B. But some component of AP1000 ECCS are safety class 3, quality group C, such as Accumulator. The design function and location of Accumulator of AP1000 is similar to that of traditional PWR.
- WEC explain:
  - Safety classification is related to the leakage probability of pressure boundary.
  - Safety class 3 can ensure the safety function.
- NRC staff thinks it is acceptable:
  - QG C is essentially equivalent to QG B, except that it has less stringent construction inspection and inservice inspection (ISI) rules.
  - All of these systems and components are located inside containment, therefore, radioactive releases are contained.
  - Minor leakage does not affect the functional performance of these systems and components ;
  - Continuous water level monitoring of the accumulators and the IRWST is performed to detect leaks.
  - the butt welds in the ECCS piping should be examined in accordance with the ASME Code, Section III, ND-5222, using the full radiography option.
- According to Policy Statement of NNSA, it is acceptable.

Selection of Accident Monitoring Parameters

- Reactor Vessel water level
  - It should be category 1 parameter according to RG1.97r3, measured from bottom of core support plate to lesser of top of vessel.
  - In AP1000 design, No direct measurement of reactor vessel water level, accomplished by using pressurizer level and hot leg level to cover the range from the bottom of the hot leg to the top of the vessel. Pressurizer level is a category 1 variable and hot leg level is a category 2 variable.
- Concentration of Hydrogen in containment
  - It should be category 1 parameter according to RG1.97r3.
  - It is category 3 parameter in the design of AP1000.
- Water level of SG secondary side
  - It should be category 1 parameter according to RG1.97r3.
  - It is category 2 parameter in the design of AP1000.

3s power supply to RCP maintained after turbine trip

- It is assumed in the accident analysis for AP1000 that at least 3s power supply to RCP can be maintained after turbine trip.
- The evaluation report of grid stability show that the grid of Zhejiang province is good enough to withstand turbine trip of NPP.
- We think it is acceptable.

The Impact of SSE to Control rod Drop Time after reactor scram

- The impact of SSE to the control rod drop time after reactor scram has not been considered in the design of AP1000, but it is conservatively considered in traditional PWR in China. It is a safety practice.
- There is no clear requirement that the impact of SSE to the control rod drop time after reactor scram should be considered in accident analysis.
- The probability of coincident of accident and SSE is extremely low, it should be considered as BDBA, it is not reasonable to consider the impact of SSE to control rod drop time after reactor scram in accident analysis.
  - Seismic induced accident should be evaluated.
- We think it is acceptable.

COL Action provided in NUREG-1793 Appendix F

- Plant Specific design of AP1000 NPPs has not been reviewed by NRC during AP1000 DC application, these plant specific design are treated as COL Action and provided in NUREG-1793 appendix F, they should be reviewed during COL application. Some important issues which can't been solved during AP1000 DC application are treated as COL Action too, such as pipe stress analysis and sump strainer design.
- In order to fully utilize NRC's review experience and review conclusion about the design of AP1000, so as to improve the efficiency of the safety review of AP1000 licensing in China, We ask the Applicant provide a list of status of COL Action items for Sanmen NPP, to describe one by one whether it is finished or not, PSAR subsection where corresponding information are provided, and corresponding technical report Number. If any COL Action items will be carried out in the future, time schedule should be provided.
- The Applicant has provided a preliminary table during PSAR stage.
- It is a CP condition.

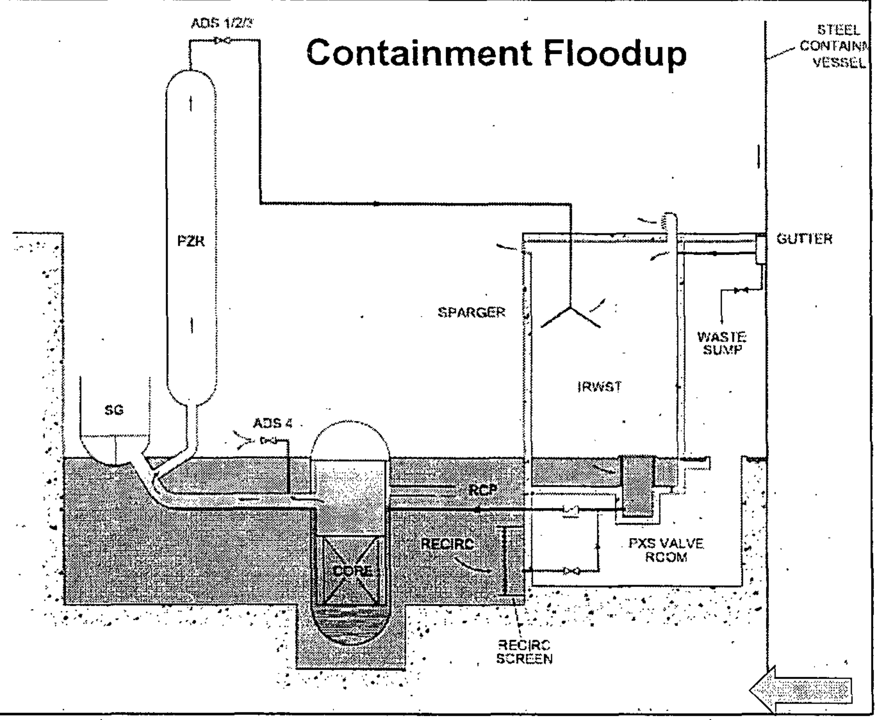
Modification of Shield Building Structure

- The structure of shield building in AP1000 DCD Revision 15 is RC, the structure of shield building in AP1000 DCD Revision 16 is changed to SC in order to consider the impact of aircraft to NI building.
- We think that the connection between common concrete structure and section steel—concrete structure is very important, but not enough information to demonstrate that it is acceptable during PSAR stage.
- The Applicant shall, within 1 month after CP, specify a time line for solving the connection issue and shall provide adequate information before the time when construction reaches such connection. The information shall illustrate the acceptability of the connection design. Regarding the theoretical basis of section steel—concrete structure used in Shield Building shell structure, the Applicant shall furnish adequate substantiating information before the construction proceeds to such concerned connection point, to demonstrate the conservatism used in this application.

Strainer Design of AP1000 sump

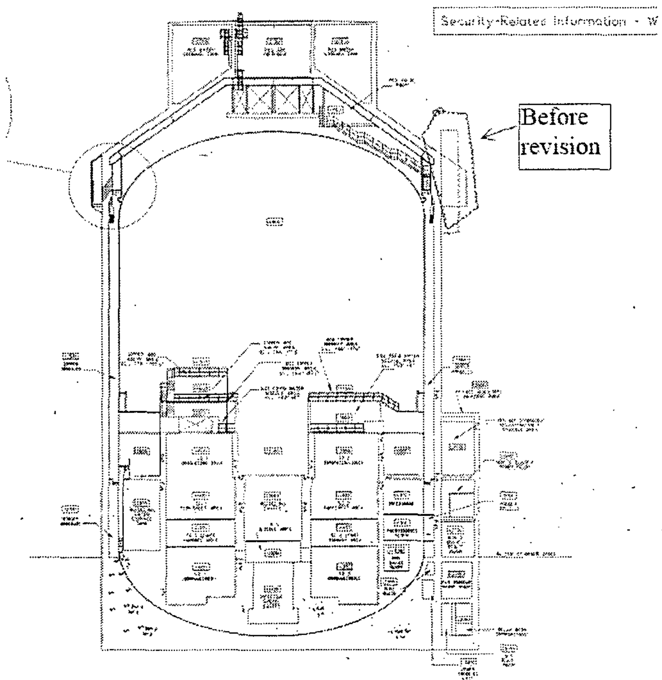
- It is GSI-191. NRC issued GL-2004-02 on Sep. 13, 2004.
- It is not solved in AP1000 DCD Revision 15. NRC treat it as COL Action.
- WEC has provide topical report, it shows that the design of strainer has been improved.
- As regards the downstream effects of sump screen, comparing AP1000 with traditional PWR design, the traditional PWR will only permit debris small than the sump screen size get into the reactor coolant system, but according to the design of AP1000, the debris larger than sump screen size (0.125") can get into the reactor coolant system through the break during and after LOCA if <u>the break is located in the RCS main pipe</u>, that will exceed the envelopment of the downstream effect analysis after LOCA for traditional PWR.
- The safety Review of this issue is going on, downstream effect and chemical product are most interesting.

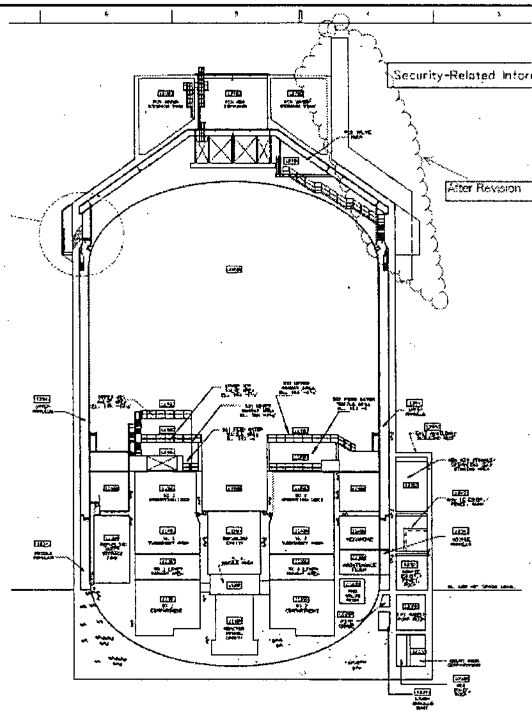
Containment Floodup



The height of the exhaust vent stack

- The height of the exhaust vent stack is lower than the height of main structure nearby, it is different from our practice.
  - GB-12697 states that the exhaust vent stack must be at least five (5) meters higher than any buildings within 200 meters in the vicinity of the stack.
- The Applicant shall, within 6 months of CP, furnish a design change proposal to change the height of the exhaust vent stack.
- Proposal of design modification by the Applicant
  - Before revision
  - After revision





Specific ITAAC for Sanmen NPP

- ITAAC is a new regulatory tools of NRC defined in 10CFR52. If the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and should operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. The first fuel loading is permitted only after all ITAAC items have been reviewed and accepted by NRC.
- Though the licensing process of NPP in China is different from that of the US 10CFR52 (similar to 10CFR50 process), however, in order to fully utilize NRC's review experience and review conclusion about the design of AP1000, so as to improve the efficiency of the safety review of AP1000 licensing in China, we will follow some parts of 10CFR52 in AP1000 licensing practice. So, we ask the Applicant provide specific ITAAC for Sanmen NPP and the schedule for completing ITAAC.

General Review Opinion

- We have not found any big issues which may overthrow the basis of issuing CP to Sanmen NPP unit 1&2 after one year of safety Review.
- Based on the information provided by the Applicant, the Policy Statement of NNSA for AP1000, and knowledge of the Reviewer nowadays, we think that PSAR of Sanmen NPP units 1&2 is acceptable.

CP Conditions (1/10)

- CP was issued to Sanmen NPP units 1&2 on March 26, 2009, with some CP conditions, such as:
  - CP5: Commensurate with the progress in AP1000 standard design change application and more in-depth power plant design, the Applicant shall, in accordance with U.S. 10CFR52 requirements, conduct the work of COL action items and ITAAC, and shall furnish a report to National Nuclear Safety Administration of China on the progress of COL action items and ITAAC, and associated work plans on a regular basis (every half year).

CP Conditions (2/10)

- CP6: Regarding the connection between common concrete structure and section steel—concrete structure, the Applicant shall, within 1 month after CP, specify a time line for solving the connection issue and shall provide adequate information before the time when construction reaches such connection. The information shall illustrate the acceptability of the connection design. Regarding the theoretical basis of section steel—concrete structure used in Shield Building shell structure, the Applicant shall furnish adequate substantiating information before the construction proceeds to such concerned connection point, to demonstrate the conservatism used in this application.

CP Conditions (3/10)

- CP8: The Applicant shall, at 2 months before the validation test of the reactor coolant pump, furnish the validation test program and test schedule for the reactor coolant pump and shall, after the completion of validation test, furnish validation test report.

CP Conditions (4/10)

- CP10-1: The Applicant shall, within half a year after CP, provide the stress analysis report for Safety Class 1 piping.

CP Conditions (5/10)

- CP10-3: The Applicant shall, within half a year after CP, further validate the appropriateness of CRDM component safety classification and seismic classification and illustrate how the expected safety functions are satisfied.

CP Conditions (6/10)

- CP10-5: The Applicant shall, within 6 months after CP, furnish a design change proposal to change the height of the exhaust vent stack.

CP Conditions (7/10)

- CP10-9: The Applicant shall, within half a year after CP, furnish further validation information, to illustrate whether adequate margin is included in rod drop time during accident analysis and whether adverse influence to rod drop time caused by various uncertainties such as measurement uncertainty has been accounted for (such as US NRC information bulletin IN-88-47).

CP Conditions (8/10)

- CP11-1: The Applicant shall, within 1 year after CP, furnish an analysis report on possible fault modes of squib valves and shall, at 3 months before planned squib validation test, furnish test program of the squib valve validation test, validation test schedule, and shall, after the completion of validation test, furnish validation test report.

CP Conditions (9/10)

- CP11-6: The annual radiation effluents discharged by Phase I project of Sanmen Nuclear Power Plant has reached a large proportion of the limit value set by Chinese national standard, therefore the Applicant shall, within 1 year, furnish a solution based on the general construction plan of later phases of Sanmen Nuclear Power Plant.

CP Conditions (10/10)

- CP12-1: For fatigue analysis, the Applicant shall refer to RG 1.207 and include the environmental impact of the reactor coolant, and within two years after CP, furnish updated fatigue analysis report.

Character of AP1000 Review Project in China

- Independent
- Full Scope
- Time limited
- NRC's Review Conclusion are referenced

Shortcoming and Disadvantage

- The efficiency of the safety review is affected due to the language problem, all of the review documents are in English;
- Some review questions has no actual progress after several Q&A rounds due to poor communication between the Reviewer and the Designer, and due to the limited number of WEC experts attended the review meeting.
- The review time is not sufficient, the Reviewer don't have enough time to read all of the review documents carefully.

Next Review Plan

- Safety Review related to CP condition.
- Some topics need to be reviewed after CP but before FSAR, so as to minimize the press of safety review of FSAR.
  - Fuel Management Strategy and its related safety assessment.
  - Realistic LB-LOCA evaluation methodology using Automated Statistical Treatment Of Uncertainty Method (ASTRUM).
  - PRA.
  - Strainer design and its related safety assessment
- Follow up the NRC's regulatory review of Application of AP1000 DC amendment and Application of COL for AP1000 in USA. Any safety related design modification of Sanmen and Haiyang should be reviewed.
- Perform audit calculation for some key design and analyses.

Progress of Engineering Projects

- Sanmen NPP FCD on Feb. 26, 2009.
- Haiyang NPP FCD on Aug. 26, 2009.
- There maybe some uncertainties during construction, equipment manufacture, commission due to lack of experience, so risk of investment (such as delay of construction and/or operation, non-conformance of important equipment, etc) maybe exist.

Design Modification Application

- Structure of Turbine Building First Bay, from steel structure to RC structure.
- Steel liners above the floor in the Auxiliary Building, adopted in the PSAR, but want to cancel it now.
- Structure of Shield Building, modified from SC to RC.

Other Issues

- Audit Calculation for Steel Plate Reinforced Concrete Structures, such as CA20.
  - JEAG 4618-2005, "Code Requirements for Nuclear Safety Related Concrete Structures".
  - ACI 349-01, "Technical Guidelines for Aseismic Design of Steel Plate Reinforced Concrete Structures - Buildings and Structures".
- NPPs located in inland.

Thanks

NNSA

The Inspection of AP1000 Construction and Manufacuring in China
AP1000建造的监督

NNSA

Shanghai Regional Office
National Nuclear Safety Administration
Mar.2010

- General Instrucion
  - 基本介绍
- Organazition
  - 组织机构
- Inspetion on AP1000
  - AP1000的监督

NNSA

Routine/non-routine inspection

- NNSA or SRO organizes some routine/non-routine inspections, which focus on some general fields (e.g. effectiveness of QA program) or some important engineering fields. By the inspection NNSA/SRO will put some requirements to licensee and make licensee to improve their performance in the field.

NNSA

Periodical dialogue meeting

- Shanghai regional office hold a dialogue meeting with licensee 3-4 times/year to discuss the safety status, existing problems and inspection schedule of plant, and propose some suggestions and requirements to licensee.
- Before meeting SRO prepares checklist and gets information and explanation from licensee. After meeting SRO prepares the memorandum of meeting and both SRO and licensee take actions to fulfill the memorandum of meeting.

NNSA

Specific inspection

- SRO chooses important construction and manufacuring activities. For every chosen item, SRO prepares the specific inspection procedure. Site-inspectors carry out the inspection according to the procedure.

NNSA

Report to NNSA

- Beweekly inspection report
- Monthly inspection report
- routine/non-routine inspection report
- Periodic dialogue minutes
- Control point inspection report
- **Specific inspection report**
- Incident review report and NCR review report
- Annual inspection report

NNSA

Inspection Activities监督实践

- Mar. 2009, NNSA organized a routine inspection focusing on preparation for FCD of Sanmen unit 1.
- Jun. 2009, NNSA organized a routine inspection focusing on effectiveness of QA program and preparation for CA20 module lifting of Sanmen unit 1.
- From Jul. 2009 to Sep. 2009, SRO organized a specific inspection focusing on CVBH construction of Sanmen unit 1.

NNSA

Inspection Activities-2

- Sep. 2009, NNSA organized a routine inspection focusing on welding and lifing preparation of CVBH of Sanmen unit 1.
- Oct. 2009, NNSA organized a routine inspection focusing on effectiveness of QA program and preparation for CVBH lifting of Sanmen Unit 1.
- Dec. 2009, NNSA organized a routine inspection focusing on preparation for NI FCD of Sanmen Unit 2.

NNSA

Inspection Activities-3

- Feb. 2010, NNSA organized a routine inspection focusing on curing NI base concrete of Sanmen unit 2.
- Mar. 2010, SRO organized a routine inspection focusing on preparation for CV first ring lifting and CA01 module lifting of Sanmen unit 1.

NNSA

Thank You!

NNSA