Roach, Edward

From: Timothy Frye OGA
Sent: Tuesday, February 03, 2009 3:35 PM
To: Eileen McKenna; Steven Schaffer; Edward Roach
Subject: FW: ACTION REQUESTED: Names of Participants for meeting with Chinese delegation February 9-11
Attachments: Chinese NNSA Feb 09 - Meeting Agenda - Rev 1.doc; ML0900702083.doc

Eileen,

Ed Roach, Steve Schaffer and I will attend the CHPB session

From: Eileen McKenna , NRO
Sent: Tuesday, February 03, 2009 2:23 PM
To: Juan Peralta; Michael Junge; Richard Laura; Timothy Frye; Anthony Hsia; Brian Thomas; David Terao; Terry Jackson; Jennifer Dixon-Herrity; Christopher Jackson; John Segala; Joseph Donoghue; Lynn Mrowca; Robert Radlinski; Charles Cox; Rebecca Karas
Cc: Terence Chan; Thomas Scarbrough
Subject: ACTION REQUESTED: Names of Participants for meeting with Chinese delegation February 9-11

Attached is the proposed agenda for the meeting. The agenda lists times for various branches to support the sessions. The first day will be at Legacy hotel, and the other days in the ACRS meeting rooms. Please respond with name(s) of who will represent your branch to me by COB Thursday February 5. Thank you all for your continued cooperation on this visit. (the second attachment is the original request with the questions)

I/18

Meeting Agenda

Chinese National Nuclear Safety Administration AP-1000 Forum

Locations: Legacy Hotel / Two White Flint Building

February 9-11, 2009

Monday, February 9 (8:30 AM – 4:30 PM) Legacy Hotel Conference Room

Breakfast (8:30 AM – 9:00 AM)

Morning Session 1 (9:00 AM – 10:30 AM)

Introductions	Jeffery Jacobson	Legacy
NRC Presentation on Review Process & Status	NWE2	
Panel Discussion on Questions: [1-1, 1-2, 1-3, 2-4, 9-1]	NWE2	

Morning Session 2 (10:45 AM – 11:45 AM)

NNSA Presentation on Review Process & Status	NNSA	Legacy

Lunch (11:45 AM – 1:15 PM) – *Lunch Provided for NNSA

Afternoon Session 1 (1:15 PM – 2:45 PM)

Presentation on ITAAC / Construction Inspection Program	DCIP	Legacy

Afternoon Session 2 (3:00 PM – 4:30 PM)

Panel Discussion on Questions: [13-1, 13-2, 13-3, 16-4, 16-5, 16-6, 16-7]	NWE2	Legacy

Adjourn

Tuesday, February 10 (9:00 AM – 3:45 PM) Two White Flint North, Room T2B1/3

Morning Session 1 (9:00 AM – 10:30 AM) - (10:45 AM – 11:30 AM)

Engineering Mechanics Panel Discussion [3-1, 3-4, 3-5, 5-6, 6-3]	EMB	T2B1

Chapter 4 & 15 Panel Discussion [4-1 – 4-6, 15-1 – 15-6]	SRSB	T2B3

Lunch (11:30 AM – 1:00 PM)

Afternoon Session 1 (1:00 PM – 2:15 PM)

Siting, Ventilation & Dose Panel Discussion [2-1, 2-2, 2-13, 6-4, 6-7, 6-8, 9-2, 15-7, 15-8]	SPCV & RSAC	T2B1

I&C and Electrical Engineering Panel Discussion [3-7, 7-1, 7-2, 15-5]	ICE & EEB	T2B3

Afternoon Session 2 (2:30 PM – 3:45 PM)

Chapter 2 Panel Discussion [2-3, 2-6, 2-8, 2-9, 2-10, 2-11, 2-12, 3-6]	RGS & SEB	T2B1

Health Physics/Tech Specs/Human-System Interface [11-1 – 11-3, 12-1, 12-2, 16-1 – 16-3, 18-1, 18-2]	COLP, CHPB & CTSB	T2B3

Adjourn

Wednesday, February 11 (9:00 AM – 2:15 PM) Two White Flint North, Room T2B1/3

Morning Session 1 (9:00 AM – 10:15 AM)

Fire Protection Panel Discussion [9-3 – 9-12]	SFPB	T2B1

Component Integrity Panel Discussion [3-2, 3-3, 3-8, 5-1 – 5-9, 6-5]	CIB	T2B3

Morning Session 2 (10:30 AM – 11:45 AM)

Quality & Maintenance Panel Discussion [1-4, 1-5, 3-6, 3-7, 5-2, 14-2, 14-3, 16-3]	SPLA & CQVP	T2B1

Sump & Rod Drop Panel Discussion [6-2, 6-5, 6-6, 14-3, 15-2]	SRSB & SPCV	T2B3

Lunch (11:45 AM – 1:15 PM)

Afternoon Session (1:15 PM – 2:15 PM)

Closing Remarks	NWE2/NNSA	T2B3

Adjourn

January 8, 2009

MEMORANDUM TO: Those On Attached List

FROM: Eileen McKenna, Branch Chief
AP1000 Projects Branch 2 **/RA/**
Division of New Reactor Licensing
Office of New Reactors

SUBJECT: SUPPORT FOR CHINA QUESTIONS AND FORTHCOMING MEETING WITH THE CHINESE NATIONAL NUCLEAR SAFETY ADMINISTRATION STAFF (TAC Q00214)

The Chinese National Nuclear Safety Administration (NNSA) is evaluating potential issuance in the near future of a construction permit for an AP1000 at the Sanmen site. During the week of February 9, 2009, a large delegation will be at NRC for a bi-lateral review meeting with the NRC. The NRC has committed to bilateral support of the Chinese and your cooperation is needed. The goal is to provide support with as little impact on staff resources as possible.

In anticipation of that meeting, they have prepared a set of questions they wish to discuss (Enclosure 1). For each question, I have made an initial assignment based upon Standard Review Plan (SRP) section and subject matter, as indicated on the enclosed list. If the question appears to be generic or process-oriented, I have assigned NWE2 as the lead. In a few instances, two branches are listed as it appears both may have a part of the response.

The specific actions requested are:

1. For questions in your assigned review areas

   a. Please confirm that appropriate branch assignment was made, and if not, advise me of proposed reassignment by January 13.

   b. Review the question and provide an answer, if possible, with a level of effort of no more than 2 hours per question. Responses suitable for transmittal to the Chinese are requested by January 23, 2009. An e-mail response is acceptable, to me with a copy to Patrick Donnelly. To the extent possible, refer to existing documentation (FSER (NUREG-1793), SRP etc). In many cases, the question relates to the certified design, not to material being amended, so the response would likely refer to the FSER. In a few instances, the actual question is hard to discern, so please advise if you cannot answer a question for this reason. Finally, some of the questions relate to areas that are still under review, in which case your response can so indicate.

   c. If a particular question cannot be answered with this level of effort, please let me know so we can discuss options.

2. Support for meeting

Depending upon responses to questions, which NNSA staff are able to attend and timing constraints, some of your staff may be asked to participate in technical meetings with NNSA for a few hours on the subjects covered by these questions. A specific schedule for each chapter discussion will be developed at a later date.

CONTACT: Eileen McKenna, NRO/DNRL
301-415-7110

Docket No. 52-006

Enclosures:
As stated

2. Support for meeting

Depending upon responses to questions, which NNSA staff are able to attend and timing constraints, some of your staff may be asked to participate in technical meetings with NNSA for a few hours on the subjects covered by these questions. A specific schedule for each chapter discussion will be developed at a later date.

CONTACT: Eileen McKenna, NRO/DNRL
301-415-7110

Docket No. 52-006

Enclosures:
As stated

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OFFICE	LA:NWE2/DNRL	BC: NWE2/DNRL
NAME	RButler	EMckenna
DATE	01/ 07/09	01/08/09

OFFICIAL RECORD COPY

MEMORANDUM TO THOSE ON THE ATTACHED LIST: January 8, 2009

SUBJECT: SUPPORT FOR CHINA QUESTIONS AND FORTHCOMING MEETING WITH NNSA STAFF (TAC Q00214)

	E-Mail Mail Stops
Charles Cox, Chief, Siting and Accident Consequences Branch	RidsNroDserRsac
Rebecca L. Karas, Chief, Geoscience and Geotechnical Branch 1	RidsNroDserRgs1
Juan Peralta, Chief, Quality and Vendor Branch 1	RidsNroDcipCqvp
Michael Junge, Chief, Operator Licensing and Human Performance Branch	RidsNroDcipColp
Richard Laura, Acting Chief, Technical Specification Branch	RidsNroDcipCcib
Timothy Frye, Chief, Health Physics Branch	RidsNroDcipChpb
Joseph Donoghue, Chief , Reactor Systems, Nuclear Performance and Code Review Branch	RidsNroDsraSrsb
John Segala, Chief, Balance of Plant Branch A	RidNroDsraSbpa
Christopher Jackson, Chief, Containment and Ventilation Branch A	RidsNroDsraSpcv
Robert Radlinski, Chief, Fire Protection Team	RidsNroDsraSfpb
Lynn Mrowca, Chief, PRA Licensing Operations Support and Maintenance Branch A	RidsNroDsraSpla
Brian Thomas, Chief, Structural Branch 1	RidsNroDeSeb1
David Terao, Chief, Component Integrity Branch 1	RidsNroDeCib1
Terry Jackson, Chief, Instrumentation and Controls Branch 1	RidsNroDeIce1
Anthony Hsia, Chief and Jennifer Dixon-Herrity, Chief, Engineering Mechanics Branches 1 and 2	RidsNroDeEmb1 RidsNroDeEmb2
Ronaldo Jenkins, Chief, Electrical Engineering Branch	RidsNroDeEeb

Questions from NNSA on AP1000
(General Design and Severe Accident, Mr. Chai Guohan)
1-1. Please explain the status of the safety review of AP1000 DC amend and COL application in USA. Is there any important RAI? Please explain some of these important RAI if possible.

1-2. How many COL action items provided in NUREG-1793 appendix F have been solved during AP1000 DC amend, please provide a list.

1-3. Whether the latest version of Regulatory Guides and Standard Review Plan (March, 2007) are applicable to the DC amend and COL application in USA, and whether an evaluation of the DC amend and COL application against the latest version of Regulatory Guides and Standard Review Plan (March, 2007) should be provided to NRC.

1-4. As regards In-Vessel Retention approach, what are the most important phenomena should be considered, are there any experimental data to support the analysis (Verification and Validation)?

1-5. As regards In-Vessel Retention approach, because it is a long term cooling process, how to consider following phenomena: (1)The impact of debris to the performance and reliability of IVR. (2)Potential boron precipitation and blockage of flow path.

(Chapter 2 and 3, Ms. Pang Rong)
2-1. Aircraft aviation near a NPP site is treated as an external event source to a NPP. Whether there is a criteria for the aviation height above which it is acceptable for a NPP. For example a height above 10000 meters, etc.

2-2. Please provide the safety review position related to military facilities near NPP sites.

2-3. For the NI building protected against commercial airplane hostility crashes, analysis has been conducted about the crash on the lateral walls. Please provide some information about the risk of crashes on roofs.

2-4. Please explain the safety concern of NRC about the applicability of AP1000 standard design to a site.

2-5. Please provide the consideration about the wind tunnel test results of AP600 can be applied in AP1000.

2-6. Please provide the review position of the consideration of the dynamic effect of the water contained in the passive containment cooling water storage tank (PCCWST), in containment refueling water storage tank (IRWST), and spent fuel pool.

2-7. Please provide the review position of the consideration the replacement of SG in the design of steel containment vessel (SCV).

ENCLOSURE 1

2-8. Please provide the review position about the determination of the value for margin-level earthquake for NPPs.

2-9. Please introduce the review process about the slide and uplift effect of AP1000 standard design NI foundation under SSE and margin earthquake.

2-10. Please introduce the consideration about the safety review of seismic analysis input parameters of AP1000 standard design.

2-11. Please provide the safety review emphases about the mass concentrating process during the seismic analysis.

2-12. Please provide suggestion about the review of the rationality of modal analysis results of 3D complex model.

2-13. In DCD chapter 2, about air temperature, maximum and minimum safety values are based on historical data and exclude peaks of less than 2 hours duration and maximum and minimum normal values are the 1 percent exceedance magnitudes. We want to know the request of NRC for time period of collecting the historical data□10 years? 30years? Or longer? And the applicant will modify the envelope value in DCD 17, NRC has accepted it or not.

(Chapter 3, Mr. Sun Zaozhan)
3-1. About the classification between high and moderate energy pipings
Sanmen NPP PSAR Subsection 3.6.1.1, on Page 3.6-4, says that "Piping systems that exceed 200°F (93.33°C) or 275 psig (1.896 MPa gauge) for two percent or less of the time during which the system is in operation or that experience high-energy pressures or temperatures for less than one percent of the plant operation time are considered moderate-energy". This does not meet the requirements of SRP BTP 3-4. How do NRC determine the criterion in BTP 3-4, and do NRC accept the criterion of "one percent of the plant operation time"?

3-2. About the material toughness of LBB pipings
SRP 3.6.3 requires that "the piping material will not become susceptible to brittle cleavage-type failures over the full range of system operating temperatures (that is, the material is on the upper shelf of the Charpy Impact energy versus test temperature curve)", but the SER of NRC on AP1000 says, quoting from NUREG 1061, that LBB analysis for brittle materials may use fracture mechanics method than the limit load method for ductile materials. If the material is on the upper shelf of the fracture toughness curve, is there still any to do the fracture mechanics analysis?

3-3. About the limitation on fatigue failure potentials for LBB candidates
SRP 3.6.3 requires that LBB candidates shall not possible fatigue failures. How should this be evaluated, for example, should the usage factor U=0.5, U=0.8 or something else for "having a potential of fatigue failure".

3-4. About the seismic qualification of CRDS by test
As for the seismic qualification of CRDS, especially the justification of rod drop time during an earthquake, is it acceptable or not if the qualification is done by analysis, or it must be done by test?

3-5. About RG 1.207
RG 1.207 was not issued when NRC write the AP1000 safety evaluation report. But, when it was issued in 2007, NRC states that "Except in those cases in which an applicant or licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the methods described in this guide will be used in evaluating submittals in connection with applications for construction permits, standard plant design certifications, operating licenses, early site permits, and combined licenses". Are the Applicants for permits or licenses after the issuance of RG 1.207 in U.S. required to justify against the requirements of this guide?

3-6. About the modular construction of building structures, how should the quality of the walls and floors be assured? Is there any need to do nondestructive examinations?

3-7. How are the functionalities of mechanical and electrical components need to be functional under severe accidents be assessed? Please give some examples.

3-8. About the PTS analysis
In Subsection 5.3.4.6 of Sanmen NPP PSAR, the screening criterion for PTS in 10CFR50.61 is used, but the criterion is based on the risk evaluation of the old NPPs. AP1000 is newly designed and adopts a direction vessel injection that is different from the old NPPs. Is the screening criterion for PTS in 10CFR50.61 is still valid for AP1000 or not?

(Chapter 4, Mr. Li Bing)
4-1. NRC have published new version of SRP (2007). In chapter 4.2 there are new requirements for the fuel system design. How do NRC deal with these new requirements during review of AP1000 DC amend and COL application?

4-2. A new type of fuel assembly is adopted in AP1000 design. Applicant should provide evaluation report for the new assembly. Unluckily we did not get any information during review of SANMEN NPP. What has NRC concerned about during review of AP1000 fuel system design?

4-3. For new core design, flow distribution, flow induced vibration and other thermal hydraulic test should be done to validate its conservativeness. Before the availability of these test results, how does NRC make the conclusion when review the core design of AP1000?

4-4. For AP1000 design, The IFM grid span does not have a rod bow penalty. The 1.5% rod bow penalty only applies to the mixing vane grid span. This is really different from other design. Is it distinct for AP1000 or common used in other NPPs in USA? What's the NRC's review position for this analysis method?

4-5. For earlier nuclear power plant design, there is large margin between maximum peak linear powers for overpower transient and for prevention of fuel centerline melt. For AP1000 this margin is nearly zero. How do NRC evaluate this design when concerned its conservative.

4-6. Please specify the review conclusion about maximum fuel burnup for AP1000.

(Chapter 5, BINE, Ms. BaiJinhua)
5-1. NRC published a generic letter GL 95-07: "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves". To response to this letter, the AP1000 designer

should take measures to solve the problem of the susceptibility to bonnet over pressurization, pressure locking, and thermal binding so as to ensure the safety function of safety-related valves.

Please introduce what kind of valve and which valves that NRC paid attention on. Related to this issue, were there any tests performed to validate the satisfied performance of the valve? If any, please provide the detail information, if necessary, please illustrate by drawings.

5-2. According to the description of AP1000 PSAR□the explosively actuated valves are used in the ADS system, IRWST and recirculation sump. Till now, we have not gotten any associated qualification information, it is said that the valve qualification will be finished around Sept. 2010. For the qualification of these valves, are there any requirements raised by NRC?
Since the explosively actuated valves are important safety related valve, we care about the valve performance when the accident happens. Please introduce what's NRC most attention paid on such valves, what is the most important affect of this valve. Please explain which parts or component of the valve are critical for the function, and list the fail mode of these explosively actuated valve to make sure that the valve will perform the safety function successfully during or after the accident.

5-3. The applicant explains that explosively actuated valve has been successfully used in BWR in USA, the further question is that: Are explosively actuated valve used in BWR important safety related valve? If they are safety related valve, how much is the inspection test frequency? Does it also follow the requirement of ASME OM-one times every two years□for detail see ASME OM□? In AP1000, since the explosively actuated valve is important safety related valve, whether NRC raised the requirement to increase the inspection test frequency?

5-4. According to the AP1000 PSAR, the fluid from the RNS over pressure safety valve will be discharge into the steam generator compartment directly. We think this will lead to contaminate of the SG compartment. In order to avoid the peopled harmed by radiation during maintenance and inspection period and reduce the cleaning time, the associated cleaning and protection requirement should be involved in the administrative procedure. Did NRC raise any question and requirement on such issue?

5-5. About main pump unit:
1□The accumulative time of AP1000 main pump model test is set to be 500 hours, is this accord with the US NRC criteria requirement? Please indicate the criteria number and the page number.
2□Is the flywheel's 125% over speed running test reasonable, which is operated alone in air in the canned motor pump model test and product test? Whether the US criteria require the flywheel should run the 125% over speed test in the running medium together with the rotor or not?
3□ How long is the duration of the main pump unit's durability running test according to US NRC requirement?
4□ What are the test requirements of the lost of external cooling water of canned motor pump refers to US criteria?

(Chapter 5 and 6, Ms. Zhang Yue)
5-6. Please explain the reasons that the 1989 edition, 1989 addenda of ASME code is used for articles NB-3200, NB-3600, NC-3600, and ND-3600 in lieu of later editions and addenda.

5-7. The screening criterion for PTS in 10CFR50.61 is used for AP1000, the criterion is based on the risk evaluation of the old NPPs, but AP1000 is newly designed, please explain why the criterion is valid for AP1000.

5-8. For inservice inspection of class 2 and 3 components, is it required that ultrasonic examination system is qualified in accordance with requirements of ASME Section XI, Appendix VIII.

5-9. Please introduce IST and IST program of snubbers and their review requirements.

(Chapter 6, Mr. Chai Guohan)
6-1. As regards the design of PRHR of AP1000, whether can it meet the requirements requested in GL-2008-01, whether has any justification report about AP1000 design been submitted to NRC as a response to GL-2008-01.

6-2. Please explain whether the Nitrogen dissolved in the water in accumulator has been considered in the AP1000 design. How to consider the impact of the dissolved nitrogen □can release from the SI flow of the accumulator□to the performance of PRHR (especially during and after MSLB).

6-3. Comparing with the Traditional PWR design, the design function and location of Accumulators of AP1000 remain the same as that of Traditional PWR. The quality group of Accumulator of Traditional PWR is group B (ASME class 2), but the quality group of Accumulator of Sanmen NPP is group C (ASME class 3), is it acceptable?

6-4. Please introduce the design features of AP1000 to solve GSI-191, and explain the status of the NRC safety review of AP1000 in this field.

6-5. 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 the break is located in the RCS main pipe, that will exceed the envelopment of the downstream effect analysis after LOCA for traditional PWR. Whether AP1000 have provided specific downstream effect analysis report?

6-6. Comparing with the traditional PWR, during long term cooling phase of AP1000 after LOCA, there is only natural circulation flow pass through the core (vessel) in the design of AP1000, and only steam will release from ADS-4 or break. So, how to prevent boron precipitation in the AP1000 reactor vessel during long term cooling after LOCA. Are there any specific experimental data to support the analysis (Verification and Validation)?

(Subsection 6.4, Mr. Xiao Jun)
6-7. As for the VES in AP1000 NPP, when it is started, the temperature in MCR rise gradually within 72h, if there are any limits for the temperature in MCR during the period? How to judge if it is acceptable?

6-8. Which standard approved by NRC to judge if the healthy condition and breathing condition are satisfied? If other country's standard has different requirements on the item, how to make the judgement? How to confirm that the positive pressure (1/8inch water column at least) can be achieved in MCR. What is the base for the value 1/8inch water column at least? If all of the American NPP satisfy the requirement?

(Chapter 7, Mr. Wang Zhongqiu)
7-1. the design of the PAMS
As per the requirements of RG1.97, the reactor vessel level measurement and the containment hydrogen concentration measurement should be category 1 parameters, but in Sanman design (AP1000), these parameters are not category 1.
In the reply of PLQ1-729, it is mentioned, Regulatory Guide 1.97 Revision 3 states that the range for coolant inventory be from the bottom of the hot leg to the top of the reactor vessel. On AP1000 this is 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.
For the hot level measurement, we think it should be category 1, because it realizes the part function of the measurement of the reactor coolant inventory. But in the reply of the PLQ2-148, it is stated, The Critical Safety Functions (CSF) for the API000 do not rely on the reactor vessel hot leg level measurement. Therefore, it is not required to be a Category 1 variable. It is classified as Category 2 because it provides useful backup information regarding core cooling and Category 3 for reactor coolant inventory because of its even less important role for that CSF.
The rationale for why the containment hydrogen concentration is not a category 1 parameter is that the US NRC has removed the design-basis LOCA from consideration in 10 CFR 50.44 and at the same time as a result are relaxing the safety classifications of the design and qualification of the hydrogen monitoring and control systems.
If the AP1000 design of PAMS is appropriate or not

7-2. The design of ATWS
The applicant stated in the reply of PLQ2-252 that the equipment performing the ATWS function is non-seismic.
In China, the equipment performing the ATWS is seismic I, we want to know the real situation in USA, and this design is accepted or not.

(Subsection 9.4, and 9.5.1, Mr. Xiao Jun)
9-1. If the site parameters (such as meteorologic parameter, water temperature) are beyond the limits set by DCD, usually how to deal with? If NRC would request the utilities to modify the design or to demonstrate that the design is still valid under such site parameters?

9-2. In USA, during the period that HVAC could not guarantee the indoor room parameters due to the choice of outdoor design parameters, if there are any specific operation procedures adopted by NPP to deal with it?

9-3. As for the max postulated fire, if the water volume calculated by 2h multiply the max fire water rate are much less than 300000gal required by SRP9.5.1 and RG1.189, if we still should abide by the requirement?

9-4. In USA, before the fire pumps start up, if there are water tanks used to stabilize the fire water system's pressure and provide the fire water at the initial stages.

9-5. In SRP, the fire resistance of fire compartment is 3h, whereas in French standard the fire resistance of fire compartment is 1.5h at least. We suppose that American standard are more liable to non-active fire protection measures in the designing of NPP. At the same time, in American standard, the fixed fire protection measures should be set for the room which have fire load above 80000BTU/ft^2(~800 MJ/m^2), which is 2 times to the French requirement 400MJ/m^2. So we deduce that the requirement for setting fixed fire protection measures in American NPP are lower than in French NPP. Please explain the base for the value 80000BTU/ft^2(~800 MJ/m^2).

9-6. Please provide and explain the method for the calculating the parameters such as fire duration, fire temperature in Fire Hazard Analysis. And explain the applicable field of the method. If the method can be applicable to various types of nuclear facilities such as NPPs, fuel manufacturing factory, fuel reprocessing plants, and research reactors?

9-7. Please introduce the application of fire dynamic tool in USA. If there is any relating software?

9-8. Please explain in detail how to satisfy the air duct requirement "Properly hung and adequately firestopped", then it can have 1h fire resistance.

9-9. Please explain in detail what kind of the fire disposal of the redundant trains can be acceptable inside the MCR and containment, in which fire compartments are not suitable to be built.

9-10. If some margin should be considered in choosing the fire resistance of the fire compartments? 10% or more?

9-11. What kind of fire extinguishing medium are selected usually in portable extinguishers in MCR.

(Subsection 9.5.1 Fire fighting, BINE, Ms. BaiJinhua)
9-12. In NRC evaluation report 9.5.1.5.c mentioned "the location of safety-related equipment and routing of Class 1E electrical cable in separate fire zones enhances the separation of redundant safe-shutdown components", please explain what kind of measures are adopted for fire zone to guarantee this function.

9-13. In NRC evaluation report 9.5.1.5.a mentioned "openings through fire barriers for pipe conduit and cable trays are sealed with noncombustible materials to provide a fire resistance rating equal to that required by the barrier", please explain why NFPA804 in chapter 6.1.3.2 fire resistance rating of fire door is less than the fire resistance rating of fire barrier?

(Chapter 11, Mr. Wu Hao)
11-1. For the source terms, please provide:
1) The basic key parameters used for the tritium source term calculation in the primary coolant water and describe the control methods in the AP1000 design for the potential tritium contamination, such as the tritium concentration control value (maximum value and the average value) in the primary coolant water, the ventilation design in the reactor building;
2) The transient source terms;
3) The basis to determine the influence of reactor coolant corrosion products activity concentration after zinc addition;

4) The issue about Nuclide release coefficient which used in calculation the fission product activity in the reactor coolant;
5) The source terms based on 1.0 percent fuel defects, and demonstrate how the WLS and WGS have the capability to process wastes based on 1.0 percent fuel defects.

(Chapter 11, BINE, Ms. BaiJinhua)
11-2. What is the acceptable contact dose rate limit of radwaste package in American near surface disposal repository?

11-3. Chemical liquid waste and spent resins□category A, B and C□is dried and hot compacted in 160L steel drum, then the waste pallet is filled in 200L steel drum and grouted, is acceptable the waste package in American near surface disposal repository?

(Chapter 12, Mr. Chen Xiaoqiu)
12-1. According to the requirement of Chinese standard □Basic standards for protection against ionizing radiation and for the safety of radiation sources□□GB18871-2002□, for occupational exposures, the applicant should provide an individual dose constraint, a risk constraint for potential exposure, and a design target for radiation protection.
We want to know whether there are similar requirements of the radiation protection design for NPP in USA and what is the NRC position for this issue.

12-2. According to the requirements of GB18871-2002 and IAEA BSS 115, radiation area is classified into controlled area and supervised area to radiation protection management and occupational exposure control:
(1) Supervised area
(a) Registrants and licensees shall designate as a supervised area any area not already designated as a controlled area but where occupational exposure conditions need to be kept under review even though specific protection measures and safety provisions are not normally needed.
(2) Controlled area
(a) Registrants and licensees shall designate as a controlled area any area in which specific protective measures or safety provisions are or could be required for controlling normal exposures or preventing the spread of contamination during normal working conditions and preventing or limiting the extent of potential exposures
(b) In determining the boundaries of any controlled area, registrants and licensees shall take account of the magnitudes of the expected normal exposures, the likelihood and magnitude of potential exposures, and the nature and extent of the required protection and safety procedures.
(c) For large scale controlled area, if exposure and contamination level change obviously in different region, different specific protection or safety measures are required. If necessary, sub zone should be classified to facilitate management.
According to above provisions, the applicant is required to demonstrate whether radiation zones of NPP meet those requirements mentioned above. In addition, the applicant should assure that 20 mSv/a of individual dose limit for occupational exposure is not exceed according to GB18871-2002.
Because of the different dose limitation between USA and China, whether the radiation protection design (including design for confinement and shield as well as radiation zoning) should be changed?

12-3. PSAR of Sanmen NPP describes access control measures to radiation work place in the normal operation and shutdown condition. These control measures should be consistent with radiation zoning.
Whether the applicant should provide the physical control measures in Zone II and Zone III?

(Chapter 13.3 Emergency Plan, Mr. Chen Xiaoqiu)
13-1. Results of evaluation of emergency planning for evolutionary and advanced reactors show that no changes to EP requirements are warranted because the potential consequences of severe accidents associated with evolutionary and passive advanced LWRs are similar to that for current reactors.
However, The NRC staff recognizes that the industry has made a significant effort to make the evolutionary and passive advanced LWRs safer than current designs, and that changes to EP requirements may be warranted if the technical criteria for EP requirements were modified to account for the lower probability of severe accidents or the longer time period between accident initiation and release of radioactive material for most severe accidents associated with evolutionary and passive advanced LWRs.
We want to know what are the new technical criteria for EP requirements and how to demonstrate that the use of increased safety in one level of the defense-in-depth framework to justify reducing requirements in another level.
For AP1000 and HTGR reactor, under what conditions, if any, can emergency planning zone be reduced, including a reduction to the site exclusion area boundary?

(Subsection 13.6 Physical Protection, BINE, Ms. BaiJinhua)
13-2. How to determine these vital targets of physical protection system based on AP1000 Probability of Safety analysis data? Please list the vital targets (important equipment in nuclear island).

13-3. Please provide the methodology of AP1000 physical protection system validity and the weakness analysis conclusion.

(Chapter 14, Mr. Chai Guohan)
14-1. As per the requirement set forced in 10CFR52.79, the COL Applicant should provide specific ITAAC for specific NPP. In addition, as per the requirement set forced in 10CFR52.99, the Applicant should provide the schedule for completing ITAAC. Whether have Bellefonte nuclear power units 3 and 4 provided specific ITAAC?

14-2. Whether the concept of first plant only test and first three plant test is acceptable to NRC.

14-3. According to the NRC Information Notice 88-47, there are two approaches to perform control rod drop time test, the previous methodology called for the interruption of power to each individual drive mechanism, the new test methodology requires the interruption of power to the rod drive mechanisms for all rods simultaneously by means of the reactor trip breakers. The test result of later is relatively longer. Which approach is used to perform control rod drop time test for traditional PWR now in the U.S? And which approach will be used to perform control rod drop time test for AP1000 NPP?

(Chapter 15, Ms. Chen Zhaolin)

15-1. Non-safety-related systems and components assumed in design-basis accident analysis

According to the requirement of HAD102/17 and the traditional assumptions used in accident analysis, one of the rules for design-basis accident is that only safety-related systems and components can be used to mitigate the accident consequence, the mitigating actuation of non-safety-related systems and components is not credit. However, several non-safety-related backup protection systems and components are used to mitigate the consequence of design-basis accident analysis in AP1000. NRC staff concludes that it is acceptable after reviewing, because the actuation of these non-safety-related systems or components are simple and reliable and included surveillance requirements (SRs) and limiting conditions for operation (LCOs) in the technical specifications (TSs). The Chinese reviewer considers: (1) The conservative accident rules should be respected for advanced and enhanced safety AP1000 plant. (2) In the event of some specific criteria can not met, applicant should put the design improvement on the first place. (3) The use of non-safety-related systems and components must be minimum and demonstrated sufficiently.

Consulting with NRC:

(1) Please explain the acceptable rules for mitigating actuation of non-safety-related systems and components assumed in accident analysis.

(2) If other non-safety-related systems and components also have the similar condition of the systems list in table 15.0-8, for example: pressurizer spray or CVCS system, can they be used to mitigate the consequence of accident analysis?

(3) Please describe the NRC review requirements and results in licensing analysis for American plant.

15-2. The insertion time of RCCA after reactor scram is not conservative.

The insertion time of RCCA after reactor scram is an important key parameter in accident analysis. The rod drop time used in accident analysis should be conservative sufficiently and be larger than the value listed in TSs to including various uncertainties and the adverse effect of safe-shutdown earthquake (SSE). The verification must be done for each RCCA to show that the measured rod drop times are less than the time limits in TSs after each RCCA for each core reloading or each removal of reactor vessel head.

However, the rod drop time (2.47s to dashpot) used in AP1000 accident analysis is equal to the time limits in TSs, i.e. without uncertainties and without SSE, and the favorable effect of loss of coolant flow (only 2.09s) is considered. The active core height is increased to be 14ft for AP1000, however the rod drop time (2.47s) is much less than the typical values (3s) of 12 ft active core.

Please explain whether the adverse effect of seismic load on control rod drop time is required in licensing accident analysis both for old and new application of American NPPs.

Please explain whether it is acceptable that the rod drop time used in accident analysis is equal to the value list in TSs, and whether it is acceptable that the favorable effect of loss of coolant flow on rod drop time is considered. Why?

15-3. SGTR analysis

Four categories are identified for AP1000 according to ANSI18.2, however the specific frequency range for each category is not given. The report of AP1000 PRA shows that the frequency of SGTR (3.88×10^{-3}) is higher than the frequency of small LOCA (5×10^{-4}), so SGTR should not be classified as a condition IV event because a small LOCA belongs to condition III.

(1) Please explain the reasonable basis of SGTR classification.

(2) Please explain the requirement of acceptance criteria of radiological consequence if a SGTR is classified as condition III?

15-4. The application of statistical methodology to deal with various uncertainties can give a larger calculation margin comparing with traditional enveloping method, by a reducing of enveloped level. However, there are special requirements for the application of statistical method, such as the application of SRSS (square root of the sum of squares) requires that the uncertainties of each parameter can be approximated by a normal distribution, and considered to be random and independent instead of as biases or dependent.
Please describe the evaluation for following statistical methods for AP1000:
(1) SRSS method is used to calculate total Fq by combining calculation uncertainty, manufacturing tolerance and rod bow factor instead of traditional method of product of each uncertainties.
(2) SRSS method is used to determine the maximum power range neutron flux setpoint shown in table 15.0-5 instead of the typical method of being added directly. A larger measure errors (such as power distribution effects 7.8% instead of 5%) are allowable for AP1000 although the maximum flux setpoint 118% of rated power is same as other plant.
(3) The method of ASTRUM (automated statistical treatment of uncertainty method) will be used in LB LOCA in order to obtain a larger margin.

15-5. The time delay assumed between turbine trip and loss of offsite power
The assumptions of loss of offsite power (LOOP) used in AP1000 accident analysis are:
LOOP is assumed in accident analysis after turbine trip..
LOOP is not considered as a single failure, and the analysis of accident analysis is performed without changing the event category.
The design provisions of AP1000 electrical system can provide power to RCPs for a minimum 3 seconds following turbine trip. This time delay (RCPs can run normally by additional 3s) has significantly effect on the results of accident analysis by giving large benefit.
Please explain in detail what safety requirements are needed to evaluate that the design of electrical system can support 3s time delay.

15-6. The flow coastdown characteristic of AP1000 canned coolant pump is very pessimistic for core cooling. However, the accident analysis results for loss of coolant flow events in 15.3 are much optimistic: no fuel rod-in-DNB in limiting events of complete loss of forced coolant flow and locked rotor, which is much favorable than similar plant with good coastdown capability of centrifugal coolant pumps.
Please explain in detail what special assumption and method are used in accident analysis which is different from traditional Westinghouse PWR, and what requirements are given by NRC.

(Radiological consequence of Accident Analysis, Mr. Chen Xiaoqiu)
15-7. For LOCA□in assumption of source term calculation (Table 15.6.5-2), aerosol removal efficiency due to impaction in containment leakage path(s) is 0.8. This will result in reducing dose consequence of LOCA greatly. The applicant explained the dose consequence for control room would exceed the limit if not considering the aerosol removal efficiency.
We want to know the position of NRC for this issue□whether the value of 0.8 is needed to be validated by test and NRC has accepted this value.

15-8. For dose consequence for control room of LOCA□the applicant has put forward "Credit was taken for the door from the vestibule to the annex building in the calculation of the effective unfiltered inleakage to the control room." That reduced the effective unfiltered inleakage from 5 cfm to 1.5 cfm.
We want to know the position of NRC for this issue.

(Chapter 16, Mr. Zhu Lixin)
16-1. SR 3.0.3 mentioned that “Up to 24 hours or up to the limit of the specified Frequency”, what is “the limit of the specified Frequency” here□

16-2. How does NRC regulate programs presented in section 5.5?

16-3. We want to know which documents involved in operational management other than TS and programs presented in 5.5 shall be reviewed by NRC.

16-4. Technical specification is mainly used for operation, but design is main concern in section 16.2, whether should this section be put into chapter 16?

16-5. We want to know whether section 16.3 need to be reviewed.

16-6. We want to know how NRC regulates NPPs before they restart after an unplanned shutdown.

16-7. Non-site-specific risk-significant SSCs have been included in DRAP of PSAR 16.2 for AP1000 type reactor, we want to know whether site-specific SSCs should be determined and associated design work should be finished and be reviewed before NRC approving Construction Permit of a new NPP.

(Chapter 18, Mr. Mao Congji)
18-1. How NRC consider for “Background noise levels should not exceed 65 dB(A) in NUREG 0700R2”

18-2. What is NRC reviewing degree on Human-System Interface Design Guidelines?

Proposed Assignments for China Questions

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