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Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: Submittal Supporting Accelerated ABWR Review Schedule - **DFSER Confirmatory Item 1.2-1**

Dear Chet:

Enclosed is a SSAR markup and a new Appendix 1B, "Comparison of U.S. ABWR and K-6/7 Difference" addressing DFSER Confirmatory Item 1.2-1.

Please provide a copy of this transmittal to Jerry Wilson.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Alan Beard (GE)
Norman Fletcher (DOE)

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1.2 GENERAL PLANT DESCRIPTION

1.2.1 Principal Design Criteria

The principal design criteria are presented in two ways. First, they are classified as either a power generation function or a safety function. Second, they are grouped according to system. Although the distinctions between power generation or safety functions are not always clear cut and are sometimes overlapping, the functional classification facilitates safety analyses, while the grouping by system facilitates the understanding of both the system function and design.

The U.S. ABWR design is similar to the international ABWR design, which is currently being designed and built at the Kashiwazaki Kariwa Nuclear Power Generation Station, Units No. 6 and No. 7 (K-6/7), by the Tokyo Electric Power Company, Inc. Differences between the U.S. ABWR design and the K-6/7 project are summarized in Appendix 1B.

1.2.1.1 General Design Criteria

1.2.1.1.1 Power Generation Design Criteria

(1) The plant is designed to produce steam for direct use in turbine-generator unit.

(2) Heat removal systems are provided with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients.

(3) Backup heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage.

(4) The fuel cladding in conjunction with other plant systems is designed to retain integrity so that the consequences of any failures are within acceptable limits throughout the range of normal operational conditions and abnormal operational transients for the design life of the fuel.

(5) Control equipment is provided to allow the reactor to respond automatically to load changes and abnormal operational transients.

(6) Reactor power level is manually controllable.

(7) Control of the reactor is possible from a single location.

(8) Reactor controls, including alarms, are arranged to allow the operator to rapidly assess the condition of the reactor system and locate system malfunctions.

(9) Interlocks or other automatic equipment are provided as backup to procedural control to avoid conditions requiring the functioning of nuclear safety systems or engineered safety features.

(10) The station is designed for routine continuous operation whereby steam activation products, fission products, corrosion products, and coolant dissociation products are processed to remain within acceptable limits.

1.2.1.1.2 Safety Design Criteria

(1) The station design conforms to applicable codes and standards as described in Subsection 1.8.2.

(2) The station is designed, fabricated, erected, and operated in such a way that the release of radioactive material to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations, for abnormal transients; and for accidents.

(3) The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient.

(4) The reactor is designed so there is no tendency for divergent oscillation of any operating characteristic considering the interaction of the reactor with other appropriate plant systems.

(5) The design provides means by which plant operators are alerted when limits on the release of radioactive material are approached.

(6) Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered safe by plant analysis.

APPENDIX 1B

COMPARSION OF U.S. ABWR AND K-6/7 DIFFERENCES

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
1.	**General Design**		
1.1	Single unit plant	Dual unit	Some facilities shared between dual units and other site units
1.2	Seismic 0.3g SSE all soils envelope	Seismic site specific	ALWR
1.3	60 year plant life	40 year	ALWR
1.4	Ultimate heat sink maximum temperature of 95°F assumed	Maximum temperature of 85°F assumed	U.S. design supports generic site envelope
1.5	U.S. Codes and Standards	MITI Codes and Standards	NRC
1.6	ABWR Product Structure	K-6/7 Product structure	None
1.7	Grid frequency 60 Hz	Grid Frequency 50 Hz	None
1.8	Radwaste system design customized for U.S.	Standard Hitachi/Toshiba design	None
2.	**Plot Plan**		
2.1	Turbine building and turbine axis in-line with reactor building	Axis perpendicular to reactor building	ALWR/Japanese choose to address turbine missile issue entirely from a structural perspective to have a more compact site plot plan
2.2	Control building located between reactor building and turbine building	Located between dual reactor buildings	Cost minimization effort for single unit plant
a.	Control room HVAC includes dual widely separated operator selectable air intakes	Single air intake	Dual intake design results in less dose to operator in U.S. control room exposure analysis
b.	RCW HX's located in basement of control building	Dedicated HX building	U.S. layout reconfigured to reflect different site plot plan
c.	RIP MG sets located in control building	RIP MG sets located in radwaste building	Individual preference

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES (Continued)

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
2.3	Radwaste building designed for a single unit	Shared facilities on multi-unit site. K-6/7 (ABWR) share facilities with K-5 (BWR-5)	Japanese emphasis on efficiency and compact site layout
2.4	Technical support center located in service building	Not required in Japan	NRC requirement that TSC be within 2 minute walk of the main control room
2.5	Condensate storage tank (CST) in yard	Storage pool located in radwaste building	CST cannot be housed in non-seismic Category I structure
2.6	Dual unit common switchgear deleted	Common switchgear used	Single versus Dual unit plant design
3.	**Power Cycle System**		
3.1	Power cycle system design meets U.S. utility preference, with emphasis on simplicity.	Japanese emphasis is on maximum heat rate and thermal efficiency	ALWR
	a. FW pumps driven by variable speed motor	Steam driven pumps	ALWR
	b. Condensate has 4x33-1/3% pumps; no condensate booster pumps	Condensate pumps plus booster pumps; 3x50% pumps at each stage	ALWR
	c. Low pressure FW heater drains cascaded back to condenser	Pumped forward	ALWR, high pressure heater drains pumped forward in both designs
	d. Moisture separator/reheaters have 1 stage reheat	2 stage reheat	ALWR
	e. Condenser is multiple pressure	Single pressure	ALWR
	f. Condenser tubing cooling water dependent	Titanium	ALWR, requirements allow use of materials suitable for actual site cooling water conditions

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES (Continued)

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
	g. Turbine gland sealing steam extracted from main steam	Dedicated system supplies clean steam	ALWR
	h. Steam jet air ejectors has 2x100% trains	1 100% train plus 1 startup train (driven by auxiliary steam)	ALWR
	i. Condenser heat sink site dependent	Sea water	ALWR
	j. TBCW system has 2x100% pumps and HXs	3x50% pumps and HXs	ALWR
	k. Condensate polishing is two stage	Single stage	ALWR, meets water quality exposure and radwaste burial volume goals
3.2	Offgas system is GE N68 design	H/T design based on earlier GE N62 design	Individual preference
3.3	Hydrogen water chemistry integral with design	Not adopted	Desirability still under study in Japan
3.4	Provisions for Zinc addition to Feedwater	No Zinc addition	Zinc addition is optional

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES (Continued)

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
4.	**Electrical Design**		
4.1	Offsite/onsite AC power sources are the low voltage generator output breaker plus one independent offsite source plus non-safety onsite gas turbine	7 unit site with multiple offsite AC power sources	U.S. design reflects ALWR requirements (both designs include normal compliment of emergency diesel generators)
4.2	Onsite power distribution network has generator output breaker and feed from gas turbine added; startup transformers deleted	No generator output breaker or gas turbine; startup transformers used to provide feed in conventional way	AC network interface designed for respective site conditions (switching logic also modified accordingly)
4.3	Isolation of 1E from non-1E loads on low voltage ac/dc circuits	Circuit Breakers are used between 1E supplies and non-1E loads	Circuit breakers are not accepted by NRC for electrical isolation of 1E and non-1E loads
4.4	DG fuel storage is 3x100% divisionally separated tanks located underground	2x200% divisionally cross-tied tanks (per reactor unit) located above ground	K-6/7 design emphasizes compact site plot plan; cross ties allowed by less rigorous divisional separation requirements
4.5	DG start capability incorporates manual (no AC) start capability	Normal capability (AC power required)	ALWR
4.6	DG fire suppression is foam system	CO_2 system	ALWR
4.7	No PVC electrical insulation allowed	Use of PVC OK	ALWR
4.8	Non-safety chillers and coolers connectable to on-site gas turbine	Gas turbine is not required	ALWR
4.9	Separation of 1E divisions is done with 3 hour fire barriers where practically achievable.	Separation of 1E divisions may utilize distance without intervening barriers.	NRC

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES (Continued)

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
5.	**Primary Containment**		
5.1	Severe accident design features	Not part of design	Subject of severe accident mitigation is still under study in Japan
	a. Containment overpressure protection	Not part of design	Passive venting of wetwell airspace through two rupture discs in series in hardened path; containment integrity recoverable by closing normally open AOVs
	b. Strengthened drywell head	Not part of design	Drywell head thickness increased from 1" to 1.25"; Pressure capability increased to near ultimate strength of balance of the containment structure
	c. Limestone concrete prohibited in lower drywell area	Not part of design	Reduces non-condensable gas generation from potential core-concrete interaction
	d. Lower drywell flooder	Not part of design	Utilizes fusible plugs on pipes connecting suppression pool to lower drywell
	e. AC independent water addition capability	Not part of design	Fire water system cross-tied into RHR with manually operated valves
	f. Onsite combustion turbine generator	Not part of desgn	ALWR
5.2	Wetwell/Drywell vacuum breakersare not testable	Vacuum breakers are air testable check valves	Testability removed based on PRA insight that additional failures are introduced.
5.3	SRV discharge piping in wetwell region specified as ASME Class 2 (MITI Class 3 equivalent) therefore, ISI is required	Specified as MITI Class 4 so no ISI required	NRC
5.4	RPV metal temperature sensor reduction	K-6/7 to have extra monitoring capability	ALWR/Extra monitoring capability not needed for follow-on plants

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES (Continued)

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
5.5	Bottom head drain line has AOV to provide isolation of drainage path in the event of an unisolable break in the CUW system outside containment.	Manual isolation valve on bottom head drain line	NRC concern, limits break path to above the top of active fuel
6.	**Secondary Containment**		
6.1	Redundant flammability control system (hydrogen recombiners) permanently installed	Portable skids - one skid normally installed in reactor building of each unit	For K-6/7 redundancy is provided by portability of skid in other unit's reactor building
6.2	SGTS has 4000 scfm capacity with auto negative pressure control capability. Redundant trains separated by 3-hour fire barriers.	1200 scfm capacity	Less prescriptive requirements for SGTS sizing in Japan; Increased capacity of U.S. system necessitates capability to control negative pressure to prevent excessive differential pressure on reactor building
6.3	Steam and FW lines classified non-seismic outboard of seismic interface restraint	Seismic out to turbine; no seismic interface restraint	Seismically qualified turbine building is standard Japanese practice
	a. Leak-before-break methodology used to eliminate pipe whip restraints	Conventionally analyzed and supported	Leak-before-break methodology still under study in Japan
6.4	HPCF pumps discharge check valve added	None	NRC/High pressure isolation
6.5	ECCS injection valve handwheel and improved position monitoring added	None	Improved sabotage resistance
6.6	CRD pump motor overspeed 25%	20%	U.S. Codes and Standards

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES (Continued)

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
6.7	Walls of upper 2 levels and the roof of the reactor building have been increased for tornado missile protection.	Tornado not a design requirement in Japan	NRC
7.	**Control Room**		
7.1	ARBM logic enforces OLMCPR, even in Manual mode, to prevent Rod Withdrawal Error transient	Logic does not enforce OLMCPR in manual mode; RWE transient analyzed as acceptable	ARBM enforcement of OLMCPR in all modes eliminates RWE as credible transient in U.S.; thus, analysis is not required
7.2	Automatic boron injection	Manual	NRC/Recirculation run back and ARI/FMCRD run in initiated from scram
7.3	Automatic suppression pool cooling for 72 hours	Manual	ALWR, No operator action required for 72 hours following transient
7.4	Automatic ADS after additional 8 minutes without high drywell pressure	Manual	NRC
	a. ADS includes manual inhibit switch on main control panel	Inhibit switch not provided	ADS inhibit switch required in U.S. to help mitigate ATWS
	b. Monitor solenoid continuity for ADS SRVs	Monitor solenoid continuity not provided	Improved sabotage resistance
7.5	RPS seismic trip is not an RPS input	Trip on high ground acceleration	Seismic scram trip is standard Japan practice
7.6	No RPS trip on TCV solenoid position	RPS trip on TCV solenoid position switch input	Standard Japan practice
7.7	RPV water level instrumentation reference zero at TAF for all instruments	Reference zero at TAF for fuel zone range only; all others use bottom of separator skirt for reference zero	In Japan, it was decided that least confusing solution is to retain past BWR practice (U.S. designed dictated by TMI Action Plan item)

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES (Continued)

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
7.8	Safety related RHR HXs outlet temperature monitor	Non 1E	NRC
7.9	Keylock switch on RHR discharge valve to radwaste	No keylock	ALWR
7.10	RPS trip on high suppression pool temperature.	Manual	ALWR. No operator action required for 30 minutes following a transient.
7.11	Auto power reduction on loss of feedwater heating	Manual	ALWR. No operator action required for 30 minutes following a transient.
7.12	Non-Class 1E uninteruptible power supplies provided.	Non-safety uninteruptible power supplies powered by 1E power source.	NRC
7.13	RPS trip provided for core power oscillations.	Manual	ALWR concern arising from core stability incidents
8.	**Water/Air**		
8.1	RCW has 3x50% vertical HXs (per division)	2x100% horizontal HXs (per division)	Differing configurations reflective of locational space constraints
	a. Corrosion monitoring subsystem included	Not included	ALWR
8.2	Essential HVAC has cooling coils in all 3 divisions; division C serves control room	Division C uses forced air only for reactor building loads and does not serve control room	Division C has less heat load and cooling coils not needed at actual conditions of K-site; U.S. design must support generic site envelope
	a. HVAC essential cooling water divisions A, B and C	Divisions A and B only	Generic site envelope
	b. Drain collection to radwaste or recycle to RCW	Storm drains	ALWR

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES (Continued)

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
8.3	HVAC normal cooling water system has increased size	Smaller size	U.S. system has larger capacity to accommodate generic site envelope
8.4	RCIC room dedicated sump	Shared sump with RHR 'A'	Dedicated RCIC sump provides considerable PRA benefit from flooding evaluation
8.5	Instrument air system has manual cross-tie back-up to nitrogen supply	Auto-transfer to back up nitrogen supply mode	There is a cross-tie between K-5, K-6 and K-7
8.6	Breathing air is dedicated system	Supplied by service air	ALWR
8.7	Service air filters and dryers added	No filters and dryers	ALWR
8.8	CRD purge water not heated.	CRD purge water is heated.	Japanese practice not used by US plants.
8.9	Reactor Service Water major components relocated to Ultimate heat sink intake structure and basement of control building.	Reactor Service Water major components located in heat exchanger building.	US ABWR design utilizes a separate ultimate heat sink.
9.	**Fire Protection**		
9.1	Physical fire barriers with 3 hour ratings used at all divisional boundaries outside containment high energy piping penetrations also require 3 hour fire ratings (or appropriate justification otherwise)	Some interdivisional equipment located in common areas designated as "non-fire zone". Penetrations do not require 3 hour ratings	Japanese practice allows some areas that contain safety related equipment (including of different divisions) to be subject to less strict fire protection requirements if supported by analysis showing probability or size of fire to be low
9.2	U.S. design has dedicated smoke removal mode consisting of dampers and logic	No such mode is required	U.S. requires capability to exhaust smoke and prevent migration to other divisions

COMPARISON OF U.S. ABWR AND K-6/7

DIFFERENCES (Continued)

	U.S. ABWR	K-6/7	REQUIREMENT/COMMENTS
9.3	Four SRVs controllable at Remote Shutdown Panel (RSP)	3 SRVs controllable per original design; U.S. design change still under study	Addition of 4th SRV at RSP improves results of fire PRA by factor of 10
10.	**Radiation**		
10.1	Containment leakage 0.5%/day assumed in dose analysis	0.4%/day assumed	Japanese data shows consistently less leakage than in U.S.; U.S. assumption reflects utility desire to retain margin for test
10.2	MSIV leakage 140 scfh total for all lines assumed in dose analysis	45 scfh total assumed	Historic Japanese data shows consistently less leakage than in U.S.; US. assumption reflects utility desire to retain margin for test
10.3	Reconfigure ARM and PRM systems to U.S. design	Site specific	Accommodate plant arrangement and processes