Case Number: 2009-0528-F-2.

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Folder Title:
North Korea Framework [9]

Staff Office-Individual:
Nonproliferation and Export Controls-Aoki, Steven

Original OA/ID Number:
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Withdrawal/Redaction Sheet
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DOCUMENT NO. AND TYPE	SUBJECT/TITLE	DATE	RESTRICTION
001a. memo	action, Steven Aoki to Anthony Lake re Geneva Agreement (2 pages)	10/20/1994	P1/b(1)
001b. memo	action, Anthony Lake to POTUS re Geneva Agreement (3 pages)	10/20/1994	P1/b(1)
001c. draft	agreement, framework, US-DPRK (4 pages)	10/20/1994	P1/b(1)
001d. draft	minute, confidential (3 pages)	10/20/1994	P1/b(1)
001e. draft	letter, POTUS to President Kim (DPRK) (1 page)	10/20/1994	P1/b(1)
002. letter	Prime Minister Murayama to POTUS (2 pages)	09/21/1994	P1/b(1)
003. paper	non-paper re North Korea negotiations (2 pages)	10/00/1994	P1/b(1)

COLLECTION:

Clinton Presidential Records
National Security Council
Steven Aoki (Nonproliferation and Export Controls)
OA/Box Number: 711

FOLDER TITLE:

NorthKorea - Framework [9]

2009-0528-F
kc1598

RESTRICTION CODES

Presidential Records Act - [44 U.S.C. 2204(a)]

P1 National Security Classified Information [(a)(1) of the PRA]
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C. Closed in accordance with restrictions contained in donor's deed of gift.
PRM. Personal record misfile defined in accordance with 44 U.S.C. 2201(3).
RR. Document will be reviewed upon request.

Freedom of Information Act - [5 U.S.C. 552(b)]

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AGREED FRAMEWORK BETWEEN
THE UNITED STATES OF AMERICA AND
THE DEMOCRATIC PEOPLE'S REPUBLIC OF KOREA

Geneva, October 21, 1994

Delegations of the Governments of the United States of America (U.S.) and the Democratic People's Republic of Korea (DPRK) held talks in Geneva from September 23 to October 21, 1994, to negotiate an overall resolution of the nuclear issue on the Korean Peninsula.

Both sides reaffirmed the importance of attaining the objectives contained in the August 12, 1994 Agreed Statement between the U.S. and the DPRK and upholding the principles of the June 11, 1993 Joint Statement of the U.S. and the DPRK to achieve peace and security on a nuclear-free Korean peninsula. The U.S. and the DPRK decided to take the following actions for the resolution of the nuclear issue:

I. Both sides will cooperate to replace the DPRK's graphite-moderated reactors and related facilities with light-water reactor (LWR) power plants.

1) In accordance with the October 20, 1994 letter of assurance from the U.S. President, the U.S. will undertake to make arrangements for the provision to the DPRK of a LWR project with a total generating capacity of approximately 2,000 MW(e) by a target date of 2003.

-- The U.S. will organize under its leadership an international consortium to finance and supply the LWR project to be provided to the DPRK. The U.S., representing the international consortium, will serve as the principal point of contact with the DPRK for the LWR project.

-- The U.S., representing the consortium, will make best efforts to secure the conclusion of a supply contract with the DPRK within six months of the date of this Document for the provision of the LWR project. Contract talks will begin as soon as possible after the date of this Document.

-- As necessary, the U.S. and the DPRK will conclude a bilateral agreement for cooperation in the field of peaceful uses of nuclear energy.

2) In accordance with the October 20, 1994 letter of assurance from the U.S. President, the U.S., representing the consortium, will make arrangements to offset the energy foregone due to the freeze of the DPRK's graphite-moderated reactors and related facilities, pending completion of the first LWR unit.

-- Alternative energy will be provided in the form of heavy oil for heating and electricity production.

-- Deliveries of heavy oil will begin within three months of the date of this Document and will reach a rate of 500,000 tons annually, in accordance with an agreed schedule of deliveries.

3) Upon receipt of U.S. assurances for the provision of LWR's and for arrangements for interim energy alternatives, the DPRK will freeze its graphite-moderated reactors and related facilities and will eventually dismantle these reactors and related facilities.

-- The freeze on the DPRK's graphite-moderated reactors and related facilities will be fully implemented within one month of the date of this Document. During this one-month period, and throughout the freeze, the International Atomic Energy Agency (IAEA) will be allowed to monitor this freeze, and the DPRK will provide full cooperation to the IAEA for this purpose.

-- Dismantlement of the DPRK's graphite-moderated reactors and related facilities will be completed when the LWR project is completed.

-- The U.S. and the DPRK will cooperate in finding a method to store safely the spent fuel from the 5 MW(e) experimental reactor during the construction of the LWR project, and to dispose of the fuel in a safe manner that does not involve reprocessing in the DPRK

4) As soon as possible after the date of this document U.S. and DPRK experts will hold two sets of experts talks.

-- At one set of talks, experts will discuss issues related to alternative energy and the replacement of the graphite-moderated reactor program with the LWR project.

-- At the other set of talks, experts will discuss specific arrangements for spent fuel storage and ultimate disposition.

II. The two sides will move toward full normalization of political and economic relations.

1) Within three months of the date of this Document, both sides will reduce barriers to trade and investment, including restrictions on telecommunications services and financial transactions.

2) Each side will open a liaison office in the other's capital following resolution of consular and other technical issues through expert level discussions.

3) As progress is made on issues of concern to each side, the U.S. and the DPRK will upgrade bilateral relations to the Ambassadorial level.

III. Both sides will work together for peace and security on a nuclear-free Korean peninsula.

1) The U.S. will provide formal assurances to the DPRK, against the threat or use of nuclear weapons by the U.S.

2) The DPRK will consistently take steps to implement the North-South Joint Declaration on the Denuclearization of the Korean Peninsula.

3) The DPRK will engage in North-South dialogue, as this Agreed Framework will help create an atmosphere that promotes such dialogue.

IV. Both sides will work together to strengthen the international nuclear non-proliferation regime.

1) The DPRK will remain a party to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) and will allow implementation of its safeguards agreement under the Treaty.

2) Upon conclusion of the supply contract for the provision of the LWR project, ad hoc and routine inspections will resume under the DPRK's safeguards agreement with the IAEA with respect to the facilities not subject to the freeze. Pending conclusion of the supply contract, inspections required by the IAEA for the continuity of safeguards will continue at the facilities not subject to the freeze.

3) When a significant portion of the LWR project is completed, but before delivery of key nuclear components, the DPRK will come into full compliance with its safeguards agreement with the IAEA (INFCIRC/403), including taking all steps that may be deemed necessary by the IAEA, following consultations with the Agency with regard to verifying the accuracy and completeness of the DPRK's initial report on all nuclear material in the DPRK.

NORTH KOREA

Q: Doesn't this agreement amount to a bribe to North Korea to meet its existing nonproliferation obligations?

A: -- No. Under the agreed framework, North Korea will take steps that go far beyond its requirements under the NPT.

-- It will freeze and eventually eliminate its graphite-moderated nuclear program. It will forego reprocessing and dismantle its reprocessing plant. It will allow the transfer out of countrty of the spent fuel it has produced.

-- These commitments by North Korea go beyond what's required by the NPT or IAEA safeguards. They will eliminate the major threat to our security and that of South Korea created by the very existence of the North Korean nuclear program.

-- Moreover, the LWR project, to be funded by South Korea and Japan, will open North Korea to more normal economic and political relations with the outside world. This also will help to reduce regional tensions and the risk of war.

Q: Why have you accepted such a long delay before special inspections to resolve North Korea's past NPT compliance? Isn't this letting them get away with violating the treaty?

A: -- The most important thing is that we've got the North to accept the <u>principle</u> that it must resolve concerns about past compliance, through special inspections, if necessary.

-- We've always said we could be flexible about the timing for this step. The evidence contained in the waste sites is not perishable.

-- We have carefully structured the framework so that the North receives no new nuclear technology until <u>after</u> it has resolved the IAEA's concerns about the past. The only things the North gets before this point are <u>supplies of fuel oil</u> and <u>non-nuclear components of the LWR</u>, like electrical generating equipment.

Q: Why have you made so many concessions to North Korea? With their new government on shaky ground, isn't this a time to be tough with the North?

A: -- As we've said all along we would do, we have been firm and clear on our objectives, but flexible about the timing and sequencing to meet those objectives.

-- We have made no concessions to North Korea on our key goals: no further separation of plutonium, an end to the current North Korean nuclear program, the eventual transfer of North Korea's spent fuel, full compliance by the North with its nonproliferation obligations, including resolution of concerns about the past, implementation of the North-South Denuclearization Declaration and resumption of North-South dialogue.

-- The cooperation with the North on LWRs and alternative energy envisaged under the agreed framework is not a concession, but a reasonable, carefully structured program aimed at replacing the North's existing, dangerous nuclear program and helping to bring North Korea out of its current isolation.

Q: Doesn't delaying special inspections damage the credibility of the NPT?

A: -- On the contratry, our dealings with North Korea demonstrate clearly that the international community is prepared to stand firmly behind the treaty, and to insist on compliance.

Q: Aren't some agencies dissatisfied with the agreement?

A: -- As the President said, all of his senior advisers recommended going forward with the agreed framework.

~~604~~ 604

ALFONSE M. D'AMATO
NEW YORK

United States Senate
WASHINGTON, DC 20510-3202

94 OCT 20 P1: 46

October 19, 1994

The Honorable William J. Clinton
President of the United States
The White House
1600 Pennsylvania Ave., NW
Washington, D.C. 20500

Dear Mr. President:

We write today to express our deep concern that the U.S. - North Korean negotiations in Geneva, Switzerland, have achieved an unacceptable resolution to the issues arising from North Korea's nuclear program. We strongly urge you to reconsider it and send the negotiators back to the table seeking a deal that will protect and advance U.S. interests in achieving a non-nuclear North Korea that is in full compliance with its international obligations under the NPT and its agreement with South Korea to denuclearize the Korean Peninsula.

Ambassador Gallucci returned to Geneva last week after consultations in Washington to offer additional concessions to North Korea. These concessions will reportedly result in delaying special inspections of the two suspect North Korean sites until five years from now or when the first of the light water reactors we propose to supply North Korea is seventy-five percent complete. Additionally, the accord commits the United States to ease trade restrictions on North Korea within three months, pledges to arrange for 500,000 tons of oil shipments annually to North Korea, and provides for opening a diplomatic liaison office in Pyongyang. Moreover, these concessions will allow the spent fuel rods North Korea removed from its graphite-moderated reactor to remain in North Korea under IAEA safeguards for a period of years.

These concessions essentially reverse longstanding U.S. policy concerning North Korea. We will have vitiated through substantial delay the IAEA special inspections process, leaving unresolved the question of how much plutonium the North recovered when it previously reprocessed fuel rods, and left in North Korea's hands the potential to trigger a major international crisis by reprocessing the spent fuel rods still under its control. In addition, we will have undermined the IAEA's credibility and made future special inspections of suspected violations virtually impossible by negotiating over IAEA's head and defining the terms of the special inspection regime bilaterally.

STATEMENT

After many months of negotiations, the U.S. and DPRK have today concluded an Agreed Framework, which provides for an overall resolution of the nuclear issue on the Korean Peninsula. The elements of this resolution include a step-by-step process to ultimately convert the DPRK's graphite moderated reactors to light water reactors ~~technology~~ and to bring the DPRK into compliance with its international non-proliferation commitments.

Beyond these technical details, the agreement also holds forth the promise of greater stability and prosperity in Northeast Asia. As the nuclear issue is resolved, opportunities will open for improved political and economic relations between the U.S. and DPRK and between the DPRK and other countries in the region, especially the ROK and Japan.

It is our hope that a resolution of the nuclear issue will create an atmosphere for both peaceful change and economic development in the region.

Minister Kang and I -- and our respective governments -- have worked hard to achieve agreement on this document. More hard work lies ahead to implement this complex ~~agreement.~~ framework We need to build a record, over the months and years ahead, of compliance, if we are to establish a basis for trust.

Today we have made a good beginning. We pledge our best efforts to implement the provisions of this Framework Document in the hope that it becomes one piece in a mosaic of peace and prosperity for Northeast Asia over the years ahead.

TO: Jim Pearce
FROM: Gary Samore

Bob Gallucci's signing statement.
Please clear with Tom Donilon
and Dan Poneman

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DOCUMENT NO. AND TYPE	SUBJECT/TITLE	DATE	RESTRICTION
001a. memo	action, Steven Aoki to Anthony Lake re Geneva Agreement (2 pages)	10/20/1994	P1/b(1)

COLLECTION:
Clinton Presidential Records
National Security Council
Steven Aoki (Nonproliferation and Export Controls)
OA/Box Number: 711

FOLDER TITLE:
NorthKorea - Framework [9]

2009-0528-F
kc1598

RESTRICTION CODES

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PRM. Personal record misfile defined in accordance with 44 U.S.C. 2201(3).
RR. Document will be reviewed upon request.

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DOCUMENT NO. AND TYPE	SUBJECT/TITLE	DATE	RESTRICTION
001b. memo	action, Anthony Lake to POTUS re Geneva Agreement (3 pages)	10/20/1994	P1/b(1)

COLLECTION:

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National Security Council
Steven Aoki (Nonproliferation and Export Controls)
OA/Box Number: 711

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DOCUMENT NO. AND TYPE	SUBJECT/TITLE	DATE	RESTRICTION
001c. draft	agreement, framework, US-DPRK (4 pages)	10/20/1994	P1/b(1)

COLLECTION:

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National Security Council
Steven Aoki (Nonproliferation and Export Controls)
OA/Box Number: 711

FOLDER TITLE:

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001e. draft	letter, POTUS to President Kim (DPRK) (1 page)	10/20/1994	P1/b(1)

COLLECTION:
Clinton Presidential Records
National Security Council
Steven Aoki (Nonproliferation and Export Controls)
OA/Box Number: 711

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U.S. Department of State

EAP FAX

Date: 10/21/94

TO: NSC – Dan Poneman / Steve Aoki

FAX Phone Number: 456-9181

Addressee's Phone: 456-9180

FROM: EAP/P – Ken Bailes

FAX Phone Number: 647-0996

Sender's Phone: 647-1028

NUMBER of PAGES INCLUDING COVER SHEET 2

Remarks: Attached are the President's remarks about alternative fuel. We're getting questions here from the Asian press, which we're stalling on providing a response to. Correcting any misimpressions early would be helpful to all. Guidance would be greatly appreciated.

UNCLASSIFIED ONLY

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The Federal News Reuter Transcript Service

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As more and more middle-income Americans will discover, this is a very good deal, which is a very important part of America's long-term strategy for economic health.

Unfortunately, there are those who don't support this approach and want to take us back to the days when working families couldn't afford to send their children to college. Every single one of our political opponents voted against the college loan reform plan. Most of them have no signed a contract telling us what they would do if they controlled Congress. They would give a $200 billion tax cut to the wealthiest Americans. They would explode the deficit. And to help pay for their promises, they have made a specific pledge to cut the student loan programs for three million American student borrowers every year.

Well, our contract is with the future. I don't want to go back. And I don't believe the American people will support this approach.
Ten days ago, I got a letter that shows how important this issue is. A 16-year-old boy named Artur Orkeesh (sp), who immigrated here from Poland just four years ago, attends Elk Grove School in Des Plains, Illinois -- here's what he wrote me about his dream of going to college:

"Since I came to the United States, my dream has been to attend a school like Harvard or Stanford. I rank number one in my class, but I know for a fact my parents are not going to be able to pay my tuition if I should get accepted to a good university. I'd like to know if students not as rich as others will get the opportunity to fulfill the American dream and graduate from a great university."

Well, Artur, if you're listening, I got your message, and the Individual Education Account will help you get your wish.

Before I take your questions, I'd like to say just a word about the framework with North Korea that Ambassador Gallucci signed this morning. This is a good deal for the United States. North Korea will freeze and then dismantle its nuclear program. South Korea and our other allies will be better protected. The entire world will be safer as we slow the spread of nuclear weapons.

South Korea, with support from Japan and other nations, will bear most of the cost of providing North Korea with fuel to make up for the

nuclear energy it is losing, and they will pay for an alternative power system for North Korea that will allow them to produce electricity while making it much harder for them to produce nuclear weapons.

U.S.-DPRK AGREED FRAMEWORK

-- On October 21, 1994, after sixteen months of negotiations, the United States and North Korea signed an Agreed Framework that ends the threat of nuclear proliferation on the Korean Peninsula and provides the basis for more normal relations between North Korea and the rest of the world.

-- This Agreed Framework not only serves the interests of the United States, but those of our allies, South Korea and Japan, as well. It will bring greater security to Northeast Asia and contribute to our efforts to end nuclear proliferation globally.

-- It will bring the DPRK into full compliance with its non-proliferation obligations under the Non-Proliferation Treaty (NPT). The DPRK affirms its NPT member status and commits to complying with its IAEA safeguards agreement.

-- The Agreed Framework terminates the existing North Korean nuclear program.

-- It ensures safe disposition of spent fuel now in North Korea. The DPRK will forego reprocessing, and instead will safely store and eventually ship the fuel out of country.

-- The DPRK will accept special inspections or other steps deemed necessary by the IAEA before it receives any nuclear components for a light water reactor (LWR).

-- The Agreed Framework addresses the following concerns about the North Korean nuclear program with respect to past activities, current activities and future activities.

The Past

-- The question of what North Korea did in the past can be resolved by the IAEA only if the IAEA has the access to information in sites that it needs.

-- Under the Agreed Framework, that access will be provided. The DPRK has agreed to the implementation of its full-scope safeguards agreement and whatever the IAEA deems necessary to resolve questions about past activities.

-- Implementation of that portion of the Agreed Framework takes place over a period of time, but must be completed before significant nuclear components of the first light water reactor (LWR) to be constructed in North Korea are delivered.

-- That is, before any nuclear components are delivered for the first LWR, the question of North Korea's past activities and North Korea's full compliance with its IAEA safeguards obligations will be (taken care of).

resolved

The Present

-- With respect to the present, North Korea has a 5 MW(e) reactor that produced plutonium in the past as well as the spent fuel now in a storage pond which, if reprocessed, would produce 25-30 kilograms of plutonium. North Korea also has a reprocessing facility.

-- Under the terms of the Agreed Framework, the North's current nuclear program is frozen: the 5 MW reactor will not restart; the reprocessing facility will be sealed and not operated again; the spent fuel which came from the 5 MW reactor will stay in the pond. These measures address our concerns about the potential for further separation of plutonium from spent fuel and further production of plutonium in the 5 MW reactor.

-- All these provisions will be monitored by the IAEA.

The Future

-- With respect to the future, the North Korean nuclear program includes two large gas graphite reactors now under construction, one rated at 50 MW(e) and the other at 200 MW(e). If those reactors were to be completed, they would produce hundreds of kilograms of plutonium a year.

-- including the reprocessing plant --

-- The construction of those reactors will be frozen, and along with ~~all other~~ facilities currently existing in North Korea, will be dismantled over the course of construction of the LWR project. That is to say, the 5 MW reactor, the reprocessing facility, the 50 MW and 200 MW reactors all will be dismantled. related

-- Dismantlement of the frozen graphite-moderated reactors and related facilities is paced to the LWR project, such that dismantlement will begin when the first LWR unit is completed and completed when the second LWR unit is completed. Dismantlement means disassembly or destruction so that components and equipment are no longer useful.

-- In addition, the spent fuel in the pond not only will not be reprocessed, but will be shipped out of North Korea. The shipment of spent fuel from the DPRK will begin when significant nuclear components begin to be delivered for the first LWR. The fuel must be completely shipped out of North Korea by the time the first LWR is completed.

Light-Water Reactor (LWR) Project

-- The Agreed Framework provides that in return for the freezing and dismantling of its present nuclear program, North Korea will receive assistance from the international community in achieving legitimate energy objectives.

-- The U.S. will lead an international consortium, tentatively called the Korean Energy Development Organization (KEDO), which will oversee construction of two 1,000 MW light water reactors of proliferation-resistant design in the DPRK over the next decade. These reactors will be largely financed and constructed by the Republic of Korea, with help from Japan and presumably other countries as well.

Alternative Energy

-- The Agreed Framework also provides that the energy needs of North Korea that would have been met by the frozen and dismantled nuclear reactors are to be addressed by the international community.

-- To compensate the DPRK for loss of energy production from not operating its 5 MW reactor and from abandoning the 50 MW and 200 MW reactors under construction, the consortium will provide the North 500,000 tons of heavy fuel oil annually for use in a specific power plant (50,000 tons in the first three months, and 150,000 tons in the first year of the agreement).

Political Aspects

-- The Agreed Framework will draw North Korea out of its dangerous isolation and help integrate Pyongyang into the economic and political mainstream of East Asia.

-- The U.S. and the DPRK will establish liaison offices in each other's capitals, something that will help us oversee implementation of this agreement and open a channel to deal with other issues that concern us.

-- Establishment of liaison offices will take place as issues, for example, with respect to consular activity and access, are resolved. There is also a commitment on the part of the U.S. and DPRK to move to full diplomatic relations at the ambassadorial level as other issues of concern are resolved. The Agreed Framework does not specify a time frame for these actions to take place.

-- We plan to reduce economic and financial restrictions selectively on U.S. citizens' dealings with the DPRK, in close consultation with the Congress.

-- We will provide a "negative security assurance." It would pledge us not to use nuclear weapons against North Korea as long as it remains a member in good standing of the NPT regime. (We have provided similar assurances to other signatories of the NPT).

-- This Agreed Framework attains all our goals, including the North's commitment to pursue South-North dialogue, without which there can be no permanent resolution of questions of peace and security on the Korean Peninsula. North Korea also will take steps to implement the North-South Joint Declaration on the Denuclearization of the Korean Peninsula.

-- We consulted with our allies, South Korea and Japan, at every stage of the negotiations, including frequent conversations between the President and President Kim Young Sam. Korea and Japan's strong support has been essential to the success of the talks with North Korea.

Drafted: EAP/K: RAChristenson/DBrown
10/15/94 SEKGEN 960
Cleared: T:JBarker
P:BHall
S/AL:JWit
PM:ENewsom
PA:CShelly
S/P:LDAnderson

Revised: S/AL:JPierce
10/20/94 SALGEN 179

United States Department of State

Washington, D. C. 20520

OFFICE OF AMBASSADOR AT-LARGE ROBERT L. GALLUCCI

(S/AL)

2201 "C" STREET, N.W., ROOM 6313 WASHINGTON, DC.C. 20520

UNCLASSIFIED

Date Sent: 10/20

Pages transmitted (including cover): 5

TO: Steve Aoki NSC
(Name) (Office)

FAX #: 395-1206

TEL #: 456-9181

FROM: JIM PIERCE

FAX #: (202) 647-1838

TEL#: 647-1225

SUBJECT: REVISED FACT SHEET - I'VE SENT TO GENEVA ASKING THEM TO REVIEW + CLEAR. FYI - LYNN TURK WILL SEND HIS TALKERS TOM'W AM, I CAN'T FIND THE GALLINI LWR PAPER + SHE'S NOWHERE TO BE FOUND. I'LL TRY AGAIN 10/21 AM.

Withdrawal/Redaction Marker
Clinton Library

DOCUMENT NO. AND TYPE	SUBJECT/TITLE	DATE	RESTRICTION
002. letter	Prime Minister Murayama to POTUS (2 pages)	09/21/1994	P1/b(1)

COLLECTION:

Clinton Presidential Records
National Security Council
Steven Aoki (Nonproliferation and Export Controls)
OA/Box Number: 711

FOLDER TITLE:

NorthKorea - Framework [9]

2009-0528-F
kc1598

RESTRICTION CODES

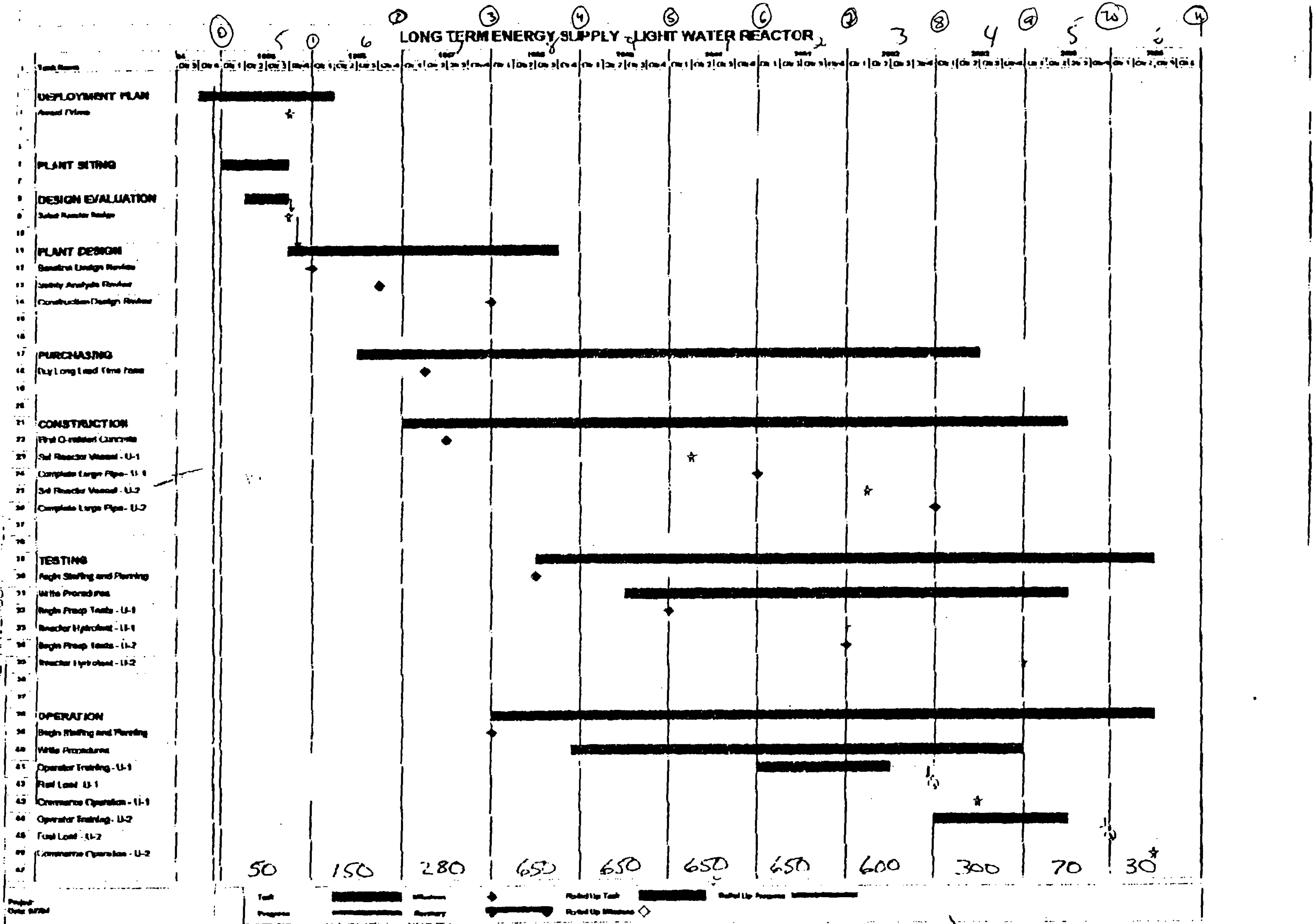
Presidential Records Act - [44 U.S.C. 2204(a)]

P1 National Security Classified Information [(a)(1) of the PRA]
P2 Relating to the appointment to Federal office [(a)(2) of the PRA]
P3 Release would violate a Federal statute [(a)(3) of the PRA]
P4 Release would disclose trade secrets or confidential commercial or financial information [(a)(4) of the PRA]
P5 Release would disclose confidential advice between the President and his advisors, or between such advisors [a)(5) of the PRA]
P6 Release would constitute a clearly unwarranted invasion of personal privacy [(a)(6) of the PRA]

C. Closed in accordance with restrictions contained in donor's deed of gift.
PRM. Personal record misfile defined in accordance with 44 U.S.C. 2201(3).
RR. Document will be reviewed upon request.

Freedom of Information Act - [5 U.S.C. 552(b)]

b(1) National security classified information [(b)(1) of the FOIA]
b(2) Release would disclose internal personnel rules and practices of an agency [(b)(2) of the FOIA]
b(3) Release would violate a Federal statute [(b)(3) of the FOIA]
b(4) Release would disclose trade secrets or confidential or financial information [(b)(4) of the FOIA]
b(6) Release would constitute a clearly unwarranted invasion of personal privacy [(b)(6) of the FOIA]
b(7) Release would disclose information compiled for law enforcement purposes [(b)(7) of the FOIA]
b(8) Release would disclose information concerning the regulation of financial institutions [(b)(8) of the FOIA]
b(9) Release would disclose geological or geophysical information concerning wells [(b)(9) of the FOIA]



200 480 1130 COST PROFILE (MILLIONS)

1780 2430 3080 3680 3980 4050 4080 Cum.

EXAMPLES OF SIGNIFICANT EQUIPMENT/MATERIALS

1. Contract Conclusion

2. Site Preparation, Infrastructure Construction and Excavation

   Construction work, roads, etc. assigned to DPRK
   Site office, security systems, storage structures, worker housing
   Communications equipment, on-site electrical supply and distribution equipment, fire station, water treatment, etc.
   Deliver structural steel and rebar
   Deliver containment vessel steel

3 Completion of initial plant design

   Drawings
   Bill of materials and labor required
   Detailed schedules for completion of reactors.

4. Specification and fabrication of major reactor components

   Detailed specifications and identification of vendors
   Fabrication of pressure vessel, steam generators, primary coolant pumps, heat exchangers, control element drive mechanism, instrumentation and control, [turbine-generator set], other equipment for future delivery to DPRK.

5. Delivery of [turbines] and turbine casings to DPRK.

$221m.

   [Turbines]
   Turbine casings
   Generators

$100 m. + installation costs

   Condenser tubes and related equipment
   Feedwater pumps and related piping
   Electrical switchgear
   Electrical controls
   Water inlet system
   Ultimate heat sink (cooling tower / cooling water discharge)

6. Construction of turbine building

$50 million labor

   Deliver building materials
   Footings/mountings
   Civil construction
   Water treatment equipment
   Electrical supply equipment, cable/wiring
   Communications and control equipment
   Wastewater treatment equipment
   Heating, ventilation and air conditioning equipment
   Bridge cranes

initial slice of US assistance to Russia for weapons dismantlement = $400 M

CAPITAL COST ESTIMATE BY EEDB COST ACCOUNT
2 UNIT

EAST/WEST CENTRAL SITE
THOUSANDS OF JANUARY, 1993 DOLLARS

ACTIVITY/ EQUIP CODE	EEDB ACCT. NUMBER	ACCT. DESCRIPTION	FACTORY EQUIPMENT	SITE LABOR HOURS	SITE LABOR	SITE MATERIAL
C	211	YARDWORK INCL. LAND COSTS		1,186,255		$2,853
	212	REACTOR BUILDING	$6,389	5,370,692		$94,264
C	213	TURBINE BUILDING	$1,486	1,663,217		$42,710
B	214	SECURITY BUILDING AND GATE HOUSE	$30	85,818		$1,302
B	215	AUXILIARY BUILDING	$21,103	7,960,337		$78,176
B	216	RADWASTE FACILITY	$1,197	1,298,701		$14,556
C	218B	ADMINISTRATION AND SERVICE BUILDING	$1,298	148,479		$3,769
B	218C	ONSITE STANDBY AC POWER GENERATION	$1,163	725,651		$7,913
B	218D	FIRE PUMPHOUSE	$59	5,392		$78
C	218K	PIPE & ELECTRIC TUNNELS		762,411		$7,669
C	218R	AUXILIARY BOILER BUILDING		1,848		$145
B	218T	ULTIMATE HEAT SINK STRUCTURE		1,554,618		$26,351
C	218Z	OTHER MISC. STRUCTURES	$668	243,996		$3,340
C	21	STRUCTURES & IMPOVEMENTS	$33,403	21,014,814		$283,228
C	222	~~MAIN HEAT TRANSPORT SYSTEM~~	$685,770	702,895		$28,821
A/B	223	~~SAFEGUARDS SYSTEM~~	$5,928	481,449		$7,651
B	224	~~RADWASTE PROCESSING~~	$4,720	194,962		$3,538
A	225	~~FUEL HANDLING & STORAGE~~	$8,088	193,914		$4,622
A	226	OTHER REACTOR PLANT EQUIPMENT	$9,366	3,004,978		$44,210
A	227	~~REACTOR PLANT INSTRUMENTATION & CONTROL~~	$1,203	42,774		$1,620
C	228	PLANT SIMULATOR	$27,172	17,000		
A	22	REACTOR PLANT EQUIPMENT	$740,247	4,617,972		$90,462
C	231	TURBINE GENERATOR	$227,091	678,231		$14,397
C	233	CONDENSING SYSTEM	$56,933	263,054		$1,545
C	234	FEED HEATING SYSTEM	$41,509	454,136		$11,270
C	235	OTHER TURBINE PLANT EQUIPMENT	$2,935	572,184		$11,376
C	237	TURBINE PLANT MISC. ITEMS	$433	63,698		$1,430
C	23	TURBINE PLANT EQUIPMENT	$328,901	2,032,303		$43,018

Note: All data based on U.S. material supply, labor costs and productivities.

CAPITAL COST ESTIMATE BY EEDB COST ACCOUNT
2 UNIT

EAST/WEST CENTRAL SITE

THOUSANDS OF JANUARY, 1993 DOLLARS

ACTIVITY EQUIP. CODE	EEDB ACCT. NUMBER	ACCT. DESCRIPTION	FACTORY EQUIPMENT	SITE LABOR HOURS	SITE LABOR	SITE MATERIAL
C	241	SWITCHGEAR *		36,078		$18,172
C	242	STATION SERVICE EQUIPMENT *	$13,305	427,353		$66,436
C	243	SWITCHBOARDS *		42,779		$8,708
C	244	PROTECTIVE EQUIPMENT *		99,233		$1,972
	245	ELECT. STRUCT. & WIRING CONTNR *		2,510,302		$7,603
C	246	POWER & CONTROL WIRING *		1,172,003		$16,559
C	24	ELECTRIC PLANT EQUIPMENT	$13,305	4,287,748		$119,448
C	251	TRANSPORTATION & LIFT EQUIPMENT *	$16,763	73,339		
C	252	AIR WATER & STEAM SERVICE SYSTEMS *	$17,593	1,189,712		$19,129
B	253	COMMUNICATION & SECURITY SYSTEM *		465,795		$7,149
C	255	WASTEWATER TREATMENT EQUIPMENT *	$160	17,749		$323
C	256	MAINTENANCE & TEST EQUIPMENT *	$7,670	1,744		$16
C	25	MISCELLANEOUS PLANT EQUIPMENT *	$42,226	1,748,339		$26,617
C	261	STRUCTURES	$3,559	868,934		$13,023
C	262	MECHANICAL EQUIPMENT *	$31,793	833,679		$3,049
C	26	MAIN COND. HEAT REJECT SYSTEM *	$35,352	1,702,613		$16,072
	31	CONTINGENCIES	$203,811	2,790,500		$67,260
		TOTAL DIRECT COSTS	$1,397,245	38,194,389		$646,105
	900	INDIRECT COSTS		11,199,234		$1,283,700
		TOTAL COSTS	$1,397,245	49,393,823		$1,929,805

Note: All data based on U.S. material supply, labor costs and productivities.

ACTIVITY/EQUIPMENT PRIORITY CODE

"A" – NSSS/NSSS RELATED

"B" – NON NSSS SAFETY EQUIPMENT

"C" – NON SAFETY/NON-NSSS BALANCE OF PLANT EQUIPMENT

* – Equipment activities that could be delivered/performed that is not NSSS regulated. [redacted]

EXAMPLES OF SIGNIFICANT EQUIPMENT/MATERIALS

gs

1. Contract Conclusion

2. Site Preparation, Infrastructure Construction and Excavation

   Construction work, roads, etc. assigned to DPRK
   Site office, security systems, storage structures, worker housing
   Communications equipment, on-site electrical supply and distribution equipment, fire station, water treatment, etc.
   Deliver structural steel and rebar
   Deliver containment vessel steel

3 Completion of initial plant design

   Drawings
   Bill of materials and labor required
   Detailed schedules for completion of reactors.

4. Specification and fabrication of major reactor components

   Detailed specifications and identification of vendors
   Fabrication of pressure vessel, steam generators, pressurizer, primary coolant pumps, heat exchangers, control element drive mechanism, instrumentation and control, [turbine-generator set], other equipment for future delivery to DPRK.

5. Delivery of [turbines] and turbine casings to DPRK.

   [Turbines]
   Turbine casings
   Generators
   Condenser tubes and related equipment
   Feedwater pumps and related piping
   Electrical switchgear
   Electrical controls
   Water inlet system

6. Construction of turbine building

   Deliver building materials
   Footings/mountings
   Civil construction
   Water treatment equipment
   Electrical supply equipment, cable/wiring
   Communications and control equipment
   Wastewater treatment equipment
   Heating, ventilation and air conditioning equipment

7. Initial reactor building/containment construction

   Pour and set concrete base mat
   Construction of reactor building
   Construction of containment
   Delivery of piping and valves for secondary cooling system
   Electrical equipment, wire/cable, communications equipment
   Air handling equipment
   Begin installation of containment steel liner
   Construct mountings for nuclear steam supply system
   Deliver inspection and quality control equipment and instruments

10/6/94

POINT PAPER

DPRK ENERGY ASSISTANCE PROGRAM
LWR Construction Sequence Evaluation

Question: What percentage of nuclear plant construction completion could be achieved before the first major piece of U.S. equipment would have to be provided under the following scenarios:

Scenario 1 - South Korean supplied NSSS with the US supplying limited proprietary components.

Scenario 2 - US supplies complete NSSS.

Scenario 1

The first component supplied by the US that would be required by a typical construction schedule would be the Reactor Coolant Pumps. These are typically set along with the other major NSSS component (eg. RPV or S/Gs) which would be at the 50% completion mark for the first unit (35% total completion for a 2 unit project) or three (3) years into the construction schedule and three (3) years from fuel load. Assuming two - three months for shipping, receipt inspections and construction preparations, the RPV could be shipped when overall construction completion is approximately 40-45% of first unit (25% total for 2 unit project), or approximately 33 months after construction has started.

Option

The point at which the installation of the reactor coolant pumps are required could be delayed until the 80% completion mark for the first unit (55% total for 2 unit project) or four and one half (4 1/2) years into the schedule. This could be accomplished if the construction sequence and schedule was planned and optimized for this scenario.

Scenario 2

The US supplies all NSSS components. Within the NSSS scope of supply, the "first" component required by the construction schedule would be the Reactor Pressure Vessel (RPV). A six year construction schedule with typical erection sequencing should require setting the RPV in place when the overall construction completion percentage is approximately 50% for the first unit (35% total for a 2 unit project), or approximately 36 months after construction has started. Assuming two - three months for shipping, receipt inspections and construction preparations, the RPV could be shipped when overall construction completion is approximately 40-45% (25% total for a 2 unit project), or approximately 33 months after construction has started.

It should be noted that if the forecasted construction completion date (fuel load) is delayed during this first 33 months of construction, delivery of the RPV can be also be delayed. With this situation, it is believed the delivery could be delayed to coincide with a time equal to 36 months prior to the revised fuel load date, with adjustments to the sequencing of work in the reactor building, without further impacting the already revised fuel load date.

Alternative:

With either scope of supply, the US could provide all the necessary equipment to support the six (6) year construction schedule and still remain in control of some key major components that would prevent the operation of the plant. These components could be held until the 85-90% completion mark for the first unit (60% total for 2 unit project) or within one and half years (1 1/2) of the scheduled fuel load. These components are as follows with the required "need" milestone:

Component	Milestone	% Complete first Unit (%total project)	Yr From FL first unit
Reactor Vessel Head,	Prior to Cold Hydro	85-90 (60)	1 1/4-1 1/2
Control Rod Drives or Reactor Protection/ Control Panels			
Fuel/Fuel Components ----	Fuel Load	100 (70)	------
Reactor Vessel Upper and Lower Internals	Prior to Hot Functional Testing	95-98 (67)	1/2

BASIS

6 Year construction schedule.

ground breaking = construction start - 0% Completion.

1st "Q" concrete pour - 6 month after construction start.

Set Major NSSS components (RPV, S/Gs RCPs) - 36 month prior to fuel load - 50% first unit completion (~35% total 2 unit completion).

Energize in-plant electrical distribution system - 24 months prior to fuel load - 75% first unit completion (~52% total 2 unit project completion).

Rx vessel & coolant system hydro - 12 months prior to fuel load ~ 95% first unit completion (~67% total 2 unit project completion).

SOUTH KOREAN CABILITIES:

South Korea has the capability to provide about 75% of the design and equipment supply for nuclear plants. Including construction labor, South Korea will do about 85% to 90% of the work on the System 80 units.

NSSS

RV Internals	ABB/CE
CRDMs	ABB/CE
RCPs	ABB/CE
Fuel Structure	ABB/CE
Electronics	ABB/CE
Reactor PV	Korea
Steam Generators	Korea
Pressurizer	Korea
Reactor Coolant Piping	Korea

BOP

AE: KOPEC/Sargent & Lundy

Turbine: Technology Transfer from GE, Korea's HANJUNG (formerly KHIC) is constructing.

Source: Regis Matzie, ABB/CE

MAJOR PROJECT MILESTONES

10/6/94

DRAFT------DRAFT-----DRAFT------DRAFT-------DRAFT------DRAFT------DRAFT------DRAFT-------DRAFT--------

ACTIVITY	TIME MONTHS	CONSTRUCTION COMPLETION PERCENTAGE			% OF FUNDS SPENT
		UNIT 1	UNIT 2	TOTAL	
Begin Site Studies	-18				
Select NSSS Prime Contractor	-15				
Select Architect/Engineer	-15				
Start Design	-12				
Award Construction Contract	-6				
Release To Mfg NSSS & Long Lead Equipment	0				
Start Site Prep	0	0	0	0	0
Pour Reactor Basemat	5	5	<5	<5	
Start Mat'l Delivery (piping, structural steel, conduit,etc.)	6	5	<5	<5	
Start Delivery of Non-sensitive equipment (pumps, valves, etc.)	12	10	<5	8	
Start U-1 Piping Install	12	10	<5	8	
Set U-1 Turbine Casing	24	25	5	20	
Start U-1 Cable Install	30	40	10	30	
Deliver U-1 NSSS Equipment	30	40	10	30	
Set U-1 NSSS Equipment	36	50	15	40	
Set U-2 Turbine Casing	36	50	15	40	
Start U-2 Piping Install	36	50	15	40	
Energize U-1 In-plant Busses	48	75	25	60	
Deliver U-2 NSSS Equipment	54	80	30	65	
Hydrotest U-1 Reactor coolant Sys.	60	95	50	80	
Set U-2 NSSS Equipment	60	95	50	80	
Start U-2 Cable Install	60	95	50	80	
Start U-1 Hot Functional Tests	64	97	60	85	
Load U-1 Fuel	72	100	65	90	
Unit 1 Commercial Operation	78		75	93	
Energize U-2 In-plant Busses	78		75	93	
Hydrotest U-2 Reactor coolant Sys.	84		95	98	
Start U-2 Hot Functional Tests	88		97	99	
Load U-2 Fuel	96		100	100	
Unit 2 Commercial Operation	102				

Combustion Engineering, Inc.

COMMITMENT KEY MILESTONE SCHEDULE

	Milestone	Unit 1 (mos. after NOA)	Unit 2 (mos. after NOA)
	Notice of Award	0	0
*	Submit PSAR to ROC-AEC for Review	9	9
*	Site Ready for NI Constructor, graded to within 1 meter of final grade; temporary utilities available	4.5	4.5
	Obtain Construction Permit from ROC-AEC	21	21
	Start Reactor Building first concrete pour	21	27
*	Energize Startup Transformer and Bus	52	64
*	Instrument Air and Service Water Systems Available	54	66
*	Submit FSAR to ROC-AEC for review	58	58
	Obtain Fuel Loading Permit from ROC-AEC and Start Pre-Core Hot Functional Tests for PWR	70	82
*	Initial Main Generator Synchronization to Grid	72	84
	Complete NI Performance/Acceptance Tests	76	88
	Declare Unit in Commercial Operation	76	88

* Requirement for Owner or Owner's A/E

Combustion Engineering, Inc.

Major Equipment/Component Delivery Schedule
(all stated in months after Notice of Award (NOA)
and are given as the time by which equipment arrives on-site)

	Unit 1 (mos. after NOA)	Unit 2 (mos. after NOA)
NSSS/Components		
Primary Reactor Components (reactor vessel and closure head, steam generators, pressurizer, hot and cold leg piping)	44	53
Reactor Coolant Pumps Internals	53	62
Reactor Vessel Internals	55	64
Control Element Drive Mechanisms	50	59
Safety Valves	54	63
Safety Injection Tanks	46	55
Discipline/Item		
C/B/A		
• Containment Vessel Steel	20	26
• Stainless Steel Liner Plate	24	30
• Structural Steel	23	29
• Extra Large Size Rebar	14	20
• Mechanical Rebar Splices	14	20

Combustion Engineering, Inc.

Major Equipment/Component Delivery Schedule
(all stated in months after Notice of Award (NOA)
and are given as the time by which equipment arrives on-site)

	Unit 1 (mos. after NOA)	Unit 2 (mos. after NOA)
Electrical		
• Electrical Containment Penetration Assemblies	37	43
• Class 1E Switchgear	42	48
• Class 1E Loadcenters	42	48
• Class 1E Motor Control Centers	42	48
• Class 1E 125v DC Equipment	39	45
• Class 1E Wire & Cable	41	47
Instrumentation & Controls		
• Simulator	48	n/a
• Nuplex 80+ Multiplexers	44	53
• Radiation Monitoring System	32	38
• Sampling System	32	38
Mechanical		
• Shutdown Cooling Heat Exchangers	22	28
• Containment & Maintenance Cranes	46	55
• Decontamination Equipment	36	42
• Essential Chilled Water Equipment	18	24

Combustion Engineering, Inc.

Major Equipment/Component Delivery Schedule
(all stated in months after Notice of Award (NOA)
and are given as the time by which equipment arrives on-site)

	Unit 1 (mos. after NOA)	Unit 2 (mos. after NOA)
• Diesel Generators	35	41
• Component Cooling Water Pumps & Exchangers	39	34

4 of 4

Target Project Schedule System 80+ Major Milestones (months after NOA)

	#1	#2
Notice of Award	0	0
PSAR	9	9
Construction Permit	21	21
1st Concrete Pour	21	27
RPV Setting	45	54
FSAR	58	58
Cold Hydro	62	74
Fuel Load	70	82
Commercial Operation	76	88

The following List of Equipment/Structures can be supplied prior to full compliance with IAEA safeguards:

Structures:

Large portion of reactor building and containment structure

Turbine building

Security buildings

Radwaste facility

Onsite power generation facility

Fire pumphouse

Auxiliary boiler building

Ultimate heat sink structure (cooling tower or water pump house)

Turbine Plant Equipment

Turbine Generator

Condensing System

Feed Heating System

Other Turbine Plant Equipment

Electric Plant Equipment

Switchgear

Station Service Equipment

Switchboards

Protective Equipment

Electric Structure and Wiring

Power and Control Wiring

Miscellaneous

Air Water and Steam Service Systems

Communication and Security Systems

Wastewater Treatement Equipment

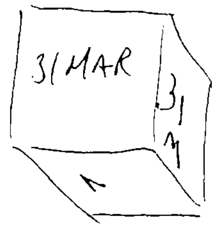
Maintenance and Test Equipment

→ Take a long time

→ refinancing w/o

→ Public Commitments

→ IAEA needs to be satisfied



Disaster for my gov't & me personally

— need resolution w/o/t having to use name

— absolutely unacceptable to us.

None of us would survive this

political

1 yR maybe 2 yrs

No way I can sell this

Korean Standard Nuclear Power Plant: safer, simpler, easier to build

By Byung Ryung Lee, Keun Sun Chang and Jun Seok Yang

KSNP — the Korean Standard Nuclear Power Plant — represents the culmination of Korea's efforts to achieve self-reliance in nuclear technology. It is intended to help the country secure a stable electric power supply for the future. Its designers also hope that it will enable the nuclear industry to continue to expand with a consensus of support from the general public. The lead KSNP units, Ulchin 3 and 4, are currently under construction.

Since the start of commercial operation of Kori 1 in 1978, a total of nine nuclear units have been put into operation in Korea — amounting to an installed capacity of 7 625MWe. Five more reactors are currently being built. There has been increasing participation by Korean organisations in design and construction activities. Table 1, showing Korean nuclear power plants in operation and under construction, reflects this. It also shows the predominance of PWRs. The PWR designs have been steadily modified and improved over the years, as indicated in Table 2, in parallel with the establishment of nuclear technologies in Korea.

The Korean standardisation programme was formulated in 1983 in direct response to the needs for a stable and reliable electricity supply in the 1990s and early in the 21st century. The programme was carried out in three distinct phases from 1983 to 1991:

- A feasibility study for a standard nuclear power plant was conducted in the first phase. In this phase, a broad spectrum of indigenous and foreign approaches to the design, construction and operation of nuclear power plants was reviewed to establish the approach and basic concept to be adopted in the Korean standard nuclear power plant.

- In the second phase, candidate designs were identified that were considered capable of being optimised and improved to meet the requirements of the standard plant. This was done by analysing the operation and construction experience of existing plants, and studying advanced technology developments in foreign countries.

- In the final phase of the programme, the complete design of the Korea Standard Nuclear Power Plant (KSNP) was developed, with Yonggwang 3 and 4 as reference plants.

The KSNP is a 1000MWe class, two-loop PWR. The design is the result of co-operative efforts between the utility, Korea Electric Power Corporation (KEPCO), the NSSS designer, Korea Atomic Energy Research Institute (KAERI), and the balance of plant designer, Korea Power Engineering Company (KOPEC).

The Korean standardisation programme culminated in a utility requirement document, K-SRED (Korea Standard Requirement Document), and in the Korea Standard Safety Analysis Report (K-SSAR), which gives a complete description of the standard design and presents detailed safety analysis for a complete nuclear power plant (except for site specific information).

Features offering significant improvements in safety, operation and construction performance have been selected and incorporated into the new design, which addresses new licensing requirements and also aims to meet the need for a reliable and stable electricity supply.

Building on proven technology, the new plant provides increased simplicity and design margin, improved man—machine interface, greater operational reliability and better constructibility than plants currently in operation.

The safety analyses indicate that there is improved response to transients and accidents and the severe accident core damage frequency is reduced by a factor of more than 10 compared with existing plants. This improvement in plant safety is mainly due to the incorporation of new accident prevention and mitigation design features, supported by the use of better materials and construction technology, simplified and balanced plant design, and other features to improve

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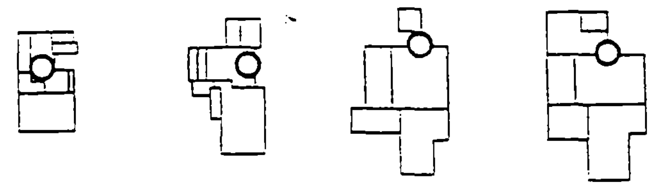
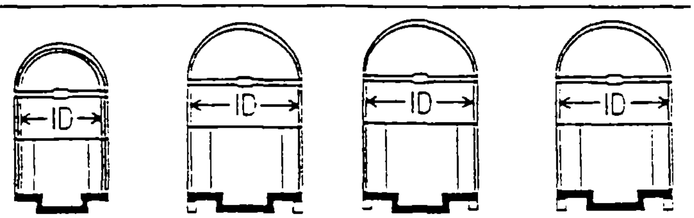
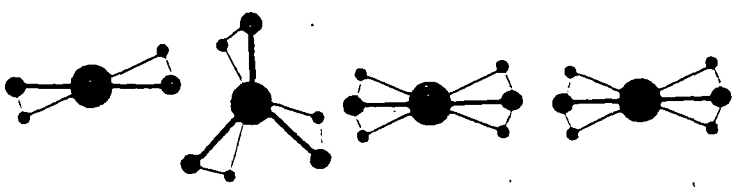
Table 1. Korean nuclear power plants in operation and under construction

Kori Nuclear Unit (No)	Name	Type	MWe	Contract	NSSS	T/G	A/E	Comm. op. date
1	Kori 1	PWR	587	Turnkey	Westinghouse	GEC	GA	1978.4
2	Kori 2	PWR	659	Turnkey	Westinghouse	GEC	GA	1983.7
3	Wolsung 1	PHWR	679	Turnkey	AECL	HP	AECL	1983.4
4	Kori 3	PWR	950	Component	Westinghouse	GEC	Bechtel	1985.9
5	Kori 4	PWR	950	Component	Westinghouse	GEC	Bechtel	1986.4
6	Yonggwang 1	PWR	950	Component	Westinghouse	Westinghouse	Bechtel	1986.8
7	Yonggwang 2	PWR	950	Component	Westinghouse	Westinghouse	Bechtel	1987.6
8	Ulchin 1	PWR	950	Component	Framatome	Alsthom	Bechtel	1988.9
9	Ulchin 2	PWR	950	Component	Framatome	Alsthom	Fram/Alst	1989.12
10	Yonggwang 3	PWR	1040	Component	KHIC/KAERI/CE	KHIC/GE	KOPEC/S&L	1995.6
11	Yonggwang 4	PWR	1040	Component	KHIC/KAERI/CE	KHIC/GE	KOPEC/S&L	1996.6
12	Wolsung 2	PHWR	700	Component	AECL/KHIC/KAERI	KHIC	AECL/KOPEC	1997.6
13	Ulchin 3	PWR	1040	Component	KHIC/KAERI	KHIC	KOPEC	1998.6
14	Ulchin 4	PWR	1040	Component	KHIC/KAERI	KHIC	KOPEC	1999.6

Table 2. Development of Korean PWRs

	Kori 1	Kori 3	Yonggwang 3	KSNP
Gross capacity (MWe)	587	950	1000	1000
Comm op. date	1978.4	1985.9	1995.3	
General layout 				
Containment building 				
Type	Steel containment	Pre-stressed concrete	Pre-stressed concrete	Pre-stressed concrete
ID (m)	32	39.6	43.9	43.9*
Design pressure (lb/in² (g))	43	60	54	57*
Free volume (m³)	41 000	61 000	76 000	76 000
Reactor coolant loops 				
Loops	2	3	2	2
No. of fuel assemblies	121	157	177	177
Reactor ID (cm)	132	157	162/396.9	162/396.9
Inlet/outlet temperature (°C)	282.6/320.2	291.9/326.9	295.8/327.3	295.8/327.3

*Subject to change due to site-specific requirements

operating effectiveness.

DESIGN APPROACH

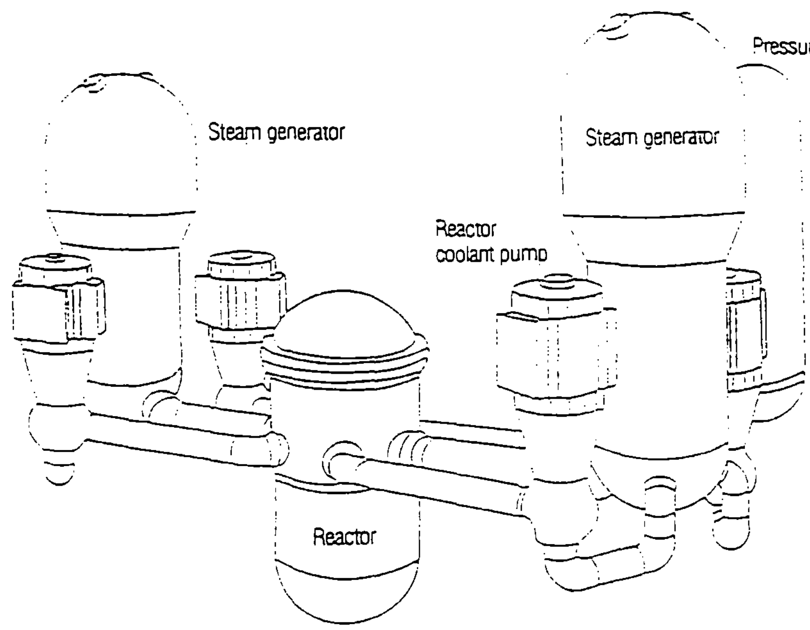
KSNP was developed through the consistent application of some fundamental design principles. These principles are compatible with Electric Power Research Institute (EPRI) advanced light water reactor (ALWR) programmes, and include the concepts of enhanced safety, simplicity, increased design margin, the use of proven technology, and more effective design and construction processes.

Simplification. Simplifications have been incorporated in all aspects of plant design, construction and operation. Plant simplifications include the use of the minimum number of systems, valves, pumps, instruments, and other items of mechanical and electrical equipment while maintaining the essential functional requirement of the related systems. The use of standardised components and modular equipment, and the separation of safety functions from non-safety systems enhance the simplification principle.

Safety policy. The assurance of licensability is a prerequisite for the successful development of KSNP. Regulatory stabilisation is achieved by resolving open licensing issues, establishing acceptable severe accident provision, and designing the plant consistent with regulatory criteria. Regulatory practice in Korea refers to that in the United States for areas that are not specified in Korean regulations. To satisfy Korean regulatory practice, a safety policy was formulated. The safety policy describes an integrated approach to safety with three overlapping levels of safety protection.

Each of these levels of safety is divided into a Licensing Design Basis and a Safety Margin Basis as depicted in Table 3. The Licensing Design Basis is the set of KSNP safety design requirements which are basically necessary to satisfy the current regulatory requirements, including licensing basis transients and accidents. The analyses for the Licensing Design Basis were done with strict, conservative, approved analysis methods and assumptions. The Safety Margin Basis contains design requirements which provide additional safety beyond the minimum required by the current regulations. A balance between accident prevention and mitigation in severe accidents has been struck in the Safety Margin Basis concept.

Design margin. The design margin provides the designed-in capability to accommodate transients without causing



▲ Fig 1. KSNP reactor system. It is a two-loop PWR.

Table 3. Korean Standard Nuclear Power Plant safety policy

	Licensing design basis (LDB)	Safety margin basis (SMB)
Accident resistance	Regulatory imposed margin, in-service inspection and testing, RCS integrity	Increased margin, simplicity, system and component reliability
Core damage prevention	Safety systems to meet regulatory requirements	Safety systems features for investment protection
	● Licensing specified accidents (mainly single failures) ● Prevent exceeding regulatory fuel limits ● RCPB integrity	● Realistic accident sequences (multiple failures) ● Greatly improved MMIS ● Prevention design features
Mitigation	Containment and associated systems	Containment performance during severe accident
	● LOCA Design Basis ● TID 14844 Source Term	● Margin beyond design basis ● Realistic source term ● Mitigation design features
Evaluation approach	Conservative, established design methods (except for a small number of multiple failure events)	Best-estimate evaluations of design margin and safety margin features
	Regulatory approved codes, standards, and acceptance criteria	Utility specified margin and acceptance criteria exceed regulatory requirements
	Credit for safety-related equipment only (except for the multiple failure events)	Credit for both safety-related and non safety-related equipment
	Deterministic licensing analyses of LDB events	Deterministic severe accident evaluations supplemented by PSA
	Meet regulations and regulatory guidance	Meet regulatory service accident policy

initiation of engineered safety systems, gives operators sufficient time to assess and deal with abnormal conditions with minimum potential for damage, and minimises the potential for exceeding limits (eg, technical specifications) which might require derating or shutdown of the plant.

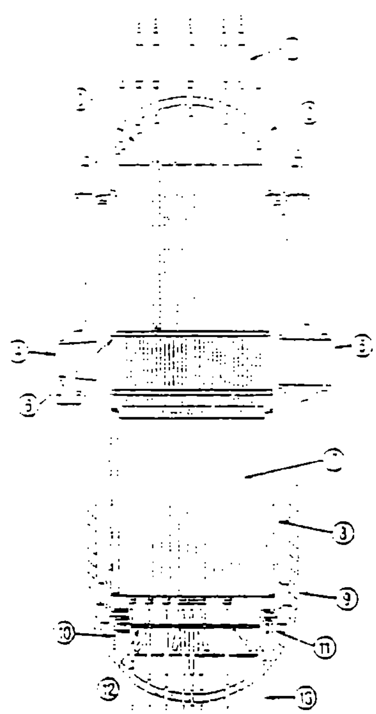
Proven technology. The KSNP design concept emphasises the employment of proven technology throughout the plant, including systems and components, maintenance and operation, and construction techniques. KSNP draws on the operating and construction experience of existing plants and the use of proven technology will help to assure stable regulation. The proven technology consists of systems, components, structures, and design and analysis techniques which have a good safety track record, as determined from the operating experience of existing plants or from regulatory review results.

Human factors. The KSNP systematically includes human factor considerations in the design of systems, facilities and equipment. The human-factors-driven design features incorporate state-of-the-art technologies in the instrumentation and control areas, including digital and microprocessor technologies. Remote multiplexers, reduced numbers of annunciators, and integrated displays, alarm procedures and controls are used in the control room design. To implement human factors principles appropriately into the design, the design process includes interaction between qualified designers and experienced operators. The use of simulators during the design process reinforces the human factors principle.

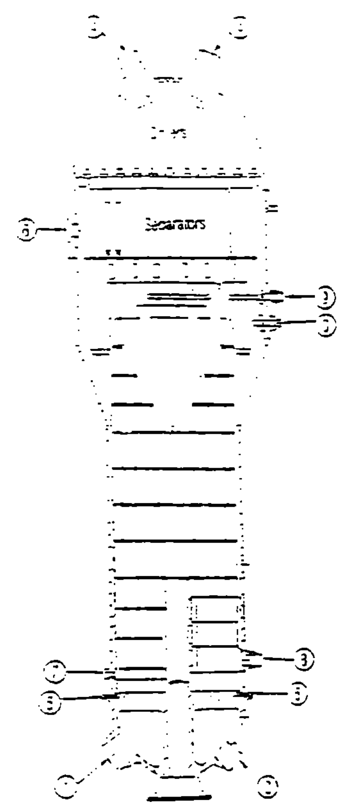
Construction and maintenance. From the outset, KSNP has been designed to be easily constructed and readily maintained. Considerations given to maintainability include providing standardised components, designing equipment to minimise maintenance needs and reduce occupational exposure, and providing more space for inspection, repair and replacement of systems and equipment. The plant has been designed and will be constructed in a single, integrated process. Provisions for modular construction, and advanced techniques of erection and installation were incorporated in the plant design at an early stage of design development.

DESIGN FEATURES: REACTOR AND RCS

The reactor is a two-loop PWR, as shown in Fig 1, with a plant design life



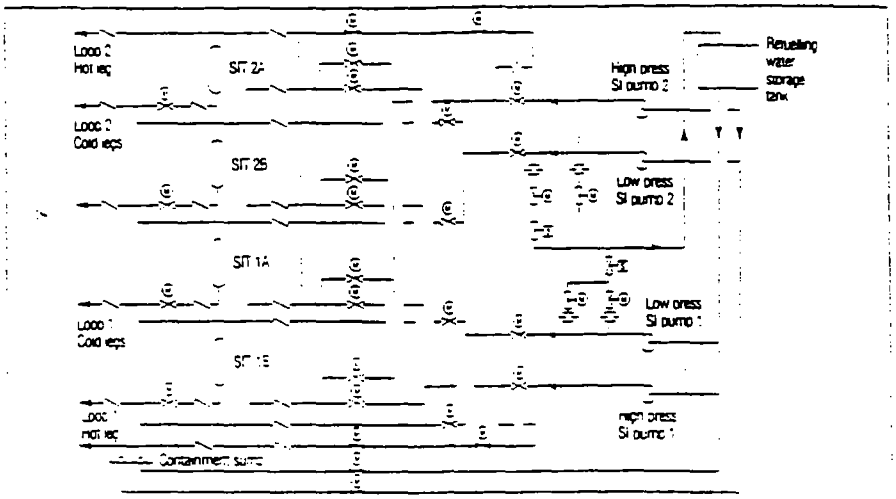
▲ Fig 2. KSNP reactor vessel and internals. 1 - CEDM. 2 - reactor vessel head assembly. 3 - CEA extension shaft. 4 - inlet nozzle. 5 - outlet nozzle. 6 - upper guide structure assembly. 7 - fuel assembly. 8 - core support barrel. 9 - reactor vessel assembly. 10 - flow skirt. 11 - core stop. 12 - lower support structure. 13 - in-core instrumentation nozzle.



▲ Fig 3. KSNP steam generator. 1 - primary inlet. 2 - primary outlet. 3 - downcomer feedwater. 4 - steam outlet. 5 - blowdown. 6 - secondary manway. 7 - handhole. 8 - economiser feedwater. 9 - recirculation.

Table 4. Characteristics of the Korean Standard Nuclear Power Plant

Reactor	
Type	PWR
Thermal output	2815 MWt
Coolant flow rate	121.5 × 10⁶ lb/h
Design pressure	2500 lb/in² (a)
Operating pressure	2500 lb/in² (a)
Design temperature	650°F
Inside diameter at shell	162 in
Overall height	48 ft
Fuel	
Number of fuel assemblies	177
Number of UO_2 fuel rods per assembly	236 (16 × 16)
Fuel weight	186 609 lb
Core height (active)	150.0 in
Core diameter (equivalent)	123.0 in
Clad material	Zircaloy-4
Clad thickness	0.025 in
Reactor coolant system	
Number of loops	2
Hot leg/cold leg	42/30 in
Reactor inlet temperature	564.5°F
Reactor outlet temperature	621.2°F
Total coolant volume	11 315 ft³
Control rods	
Number of control assemblies	73
Number of rods per assembly	4 or 12
Material (full/part strength)	B_4C/Inconel
Steam generators	
Type, number of units	Vertical U-tube, 2
Steam flow per steam generator	6.364 × 10⁶ lb/h
Steam pressure at full power	1070 lb/in² (a)
Steam temperature at full power	550.5°F
Maximum moisture	0.25%
Feedwater temperature	450°F
Reactor coolant pumps	
Number	4
Motor/type	AC induction/ vertical, centrifugal
Design capacity	82 500 gal/min
Design head	340 ft
Containment	
Type	Prestressed cylindrical concrete with steel liner
Inside diameter	144 ft
Height	216 ft
Free volume	2.73 × 10⁶ ft³
Liner thickness	0.144in
Turbine	
Number	4 (high 1, low 3)
Type	Serial 6 flow arrangement
RPM	1800
Generator	
Number, type	1, 4 poles (1800 RPM)
Voltage	22 kV, 3 phases
Frequency	60 Hz
Net electrical output	1000 MWe
Condenser	
Number, type	3, once-through sea water cooling
Pump type	Vertical, centrifugal
Pump number	50% × 3



▲ Fig 4. Safety Injection System of the KSNP.

of 40 years. The main characteristics of the KSNP design are summarized in Table 4.

Reactor The reactor vessel and internals are shown in Fig 2. Design improvements have been made for the core supervisory function and operating margin. The reactor vessel is fabricated using ring forgings. The ring-forged fabrication method, as compared with the rolled and welded plate fabrication of previous plant designs, reduces the number of welds and simplifies the vessel. The ring-forged fabrication also reduces construction time and enhances the vessel's resistance to brittle fracture since the weld areas are moved to regions of lower neutron flux.

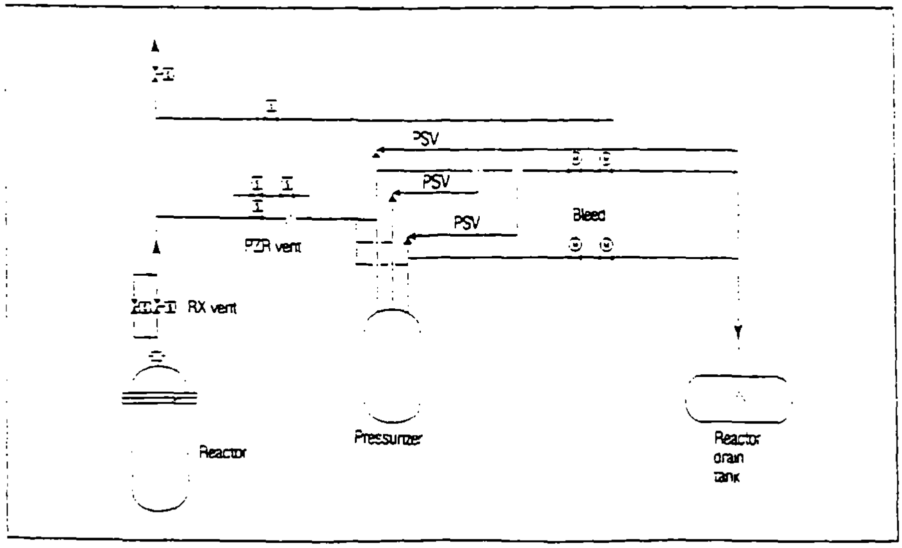
Steam generators. The steam generators, each consisting of a U-tube bundle, integral economiser and evaporator, water separators and steam dryers, are shown in Fig 3.

Improved feedwater recirculation has been achieved by optimising feedwater flow shared between the economiser and downcomer region of the steam generator. At full power, about 10% of the feedwater flow is directed to the downcomer section, and the remainder is admitted through the economiser section of the steam generator. This improvement results in a reduction of water hammer effects in the feedwater system, and an increase in thermal efficiency.

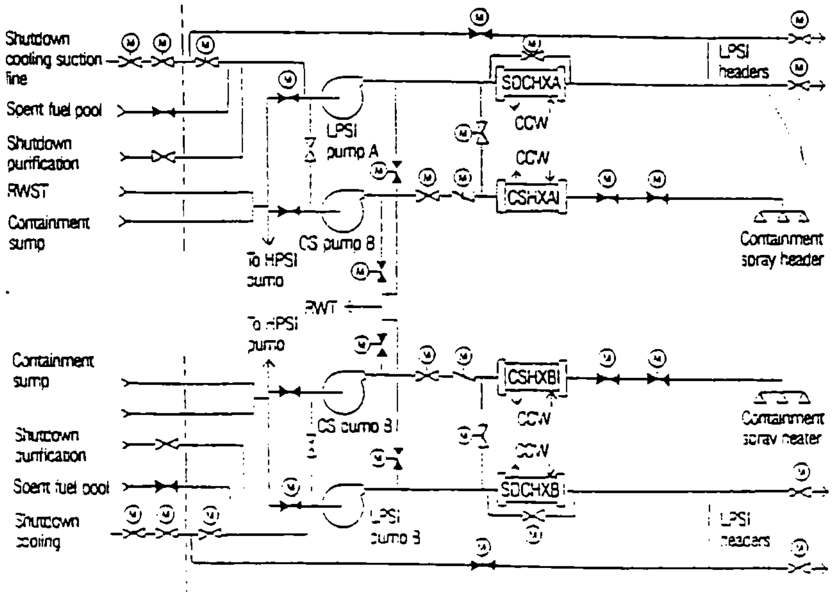
An 8% tube plugging margin in the design ensures a longer full power life for steam generators.

Pressuriser The pressuriser volume has been increased (it is 33% larger (relative to thermal output) than that found in ABB-CE designed System 80 plants.. This can significantly improve plant response to transient and accident conditions.

Also, the layout of the surge line is appropriately arranged to reduce thermally induced stress due to thermal



▲ Fig 5. KSNP Safety Depressurization and Vent System.



▲ Fig 6. KSNP Shutdown Cooling System and Containment Spray System.

stratification. A small make-up spray flow is continuously admitted to the pressuriser to avoid stratification of pressuriser boron concentration and to maintain the desired temperature in the surge and spray lines.

DESIGN FEATURES: SAFEGUARDS SYSTEMS

The KSNP safeguards systems comprise the Safety Injection System (Emergency Core Cooling System), the Safety Depressurisation and Vent System, the Shutdown Cooling System (Residual Heat Removal System), the Emergency Feedwater System and the Containment Spray System. The safeguards systems represent the area in which the most significant improvements have been made in KSNP relative to the reference plants, Yonggwang 3 and 4.

The safeguards systems are designed not only to accommodate design basis transients and accidents, but also to prevent and mitigate postulated severe accidents.

Safety Injection System. The Safety Injection System delivers borated water into the reactor coolant system to flood and cool the core following a loss of coolant accident. The Safety Injection System (Fig 4), in conjunction with the bleed function of the Safety Depressurisation and Vent System, is also designed to remove core decay heat and to restore the reactor coolant system (RCS) inventory in the event of a total loss of feedwater (feed and bleed function). The principal components of the Safety Injection System are two low pressure and two high pressure safety injection pumps, four safety injection tanks, a refuelling water storage tank, and associated piping and valves.

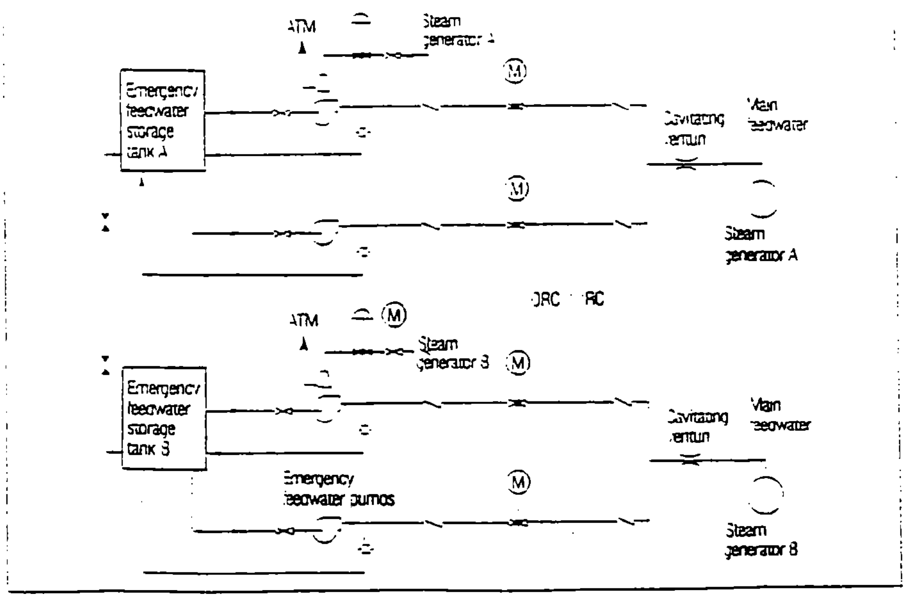
Safety Depressurisation and Vent System. The Safety Depressurisation and Vent System is designed to perform the following functions (Fig 5):

- Venting non-condensible gases from the reactor coolant system.
- Cooling the reactor coolant system when the pressuriser spray is out of order.
- Depressurising the reactor coolant system and removing heat from the reactor coolant system in the event of a beyond design basis accident consisting of total loss of feedwater.

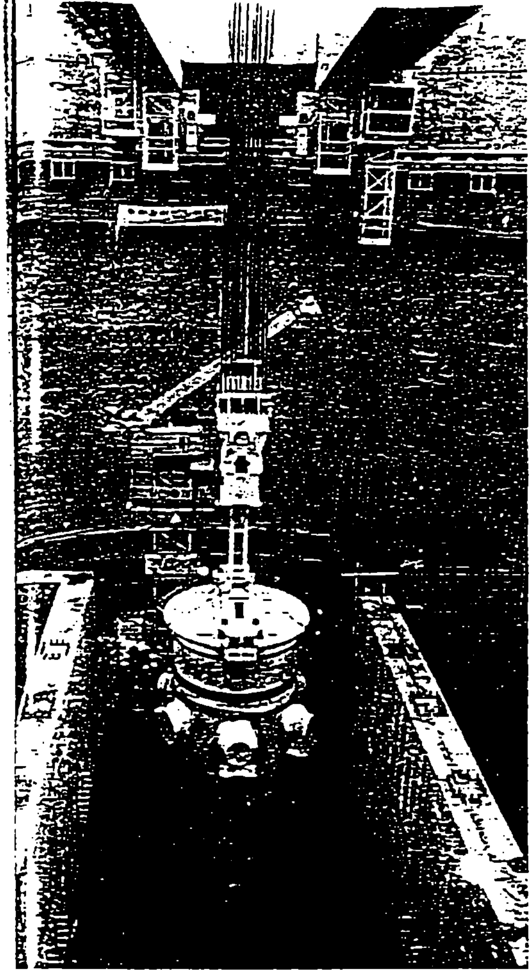
The safety depressurisation function of the Safety Depressurisation and Vent System is one of the major improvements in the KSNP design. The core damage frequency can be significantly reduced once account is taken of the feed and bleed function when the normal and emergency feedwater supplies are unavailable to remove heat through the steam generators. The bleed function is accomplished via the remote, manual opening of the motor operated valve in the bleed line of the system. As RCS pressure decreases, the Safety Injection System initiates feed flow to the RCS and restores the RCS water inventory.

The depressurisation function in conjunction with an enlarged reactor cavity arrangement can preclude the possibility of early containment failure due to ejected core debris under high pressure should the reactor vessel fail following a core melt. Other systems that help to prevent or mitigate severe accidents include an alternative AC power source for dealing with station blackout events, a large rugged dry containment, and hydrogen igniter systems for combustible gas control.

Shutdown cooling and containment spray. The Shutdown Cooling System (SCS) is used to provide a forced circulation path for decay heat removal from the RCS, and to transfer the heat to the Component Cooling Water System at RCS temperature and pressure below or equal to 350°F and 400lb/in²(a). The SCS makes use of the low pressure Safety Injection System. The design pressure of the SCS has been increased to 900lb/in²(a), compared with an SCS pressure of 485-750lb/in² for Yonggwang 3 and 4. The ultimate rupture strength of the



▲ Fig 7. Emergency Feedwater System of the KSNP.



▲ Yonggwang 3. Yonggwang 3 and 4 are the reference units for KSNP.

ing has been used to enhance operator effectiveness — enabling two operators to control the plant during normal operation.

The main control room contains a compact control board with multiple display and control devices which provide organised, hierarchical access to alarms, displays and controls.

The compact control board has the full capability to perform all main control room functions as well as support division of operator responsibilities. A supervisor's console is also located in the main control room to be used for monitoring.

The plant protection and control systems employ modern digital technology, including multiplexing and fibre optics both for monitoring and control functions. Multiplexing provides more reliable signal transfer, reduces cost and simplifies maintenance due to reduction in the complexity of cable runs. Protection and control systems are physically and logically separated between redundant systems.

DESIGN FEATURES:PSA

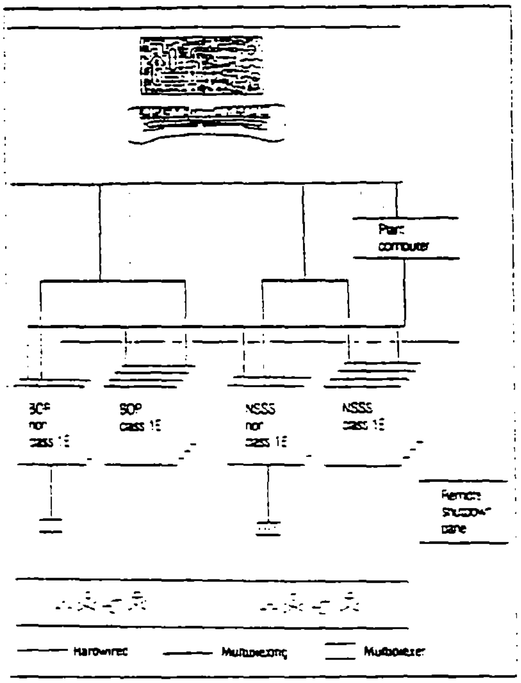
A level 1 probabilistic safety assessment (PSA) has been performed for the KSNP design. Sensitivity studies have also been performed to analyse the effectiveness of the advanced design features employed. The core damage frequency (CDF) estimated for KSNP is reduced by a factor of more than 10 compared with Yonggwang 3 and 4, the reference units.

The reduction of CDF is mainly due to the employment of advanced design features. For example, the alternate AC sources, Safety Depressurisation System and Emergency Feedwater System reduce the figure by factors of 1.9, 5.6 and 2.6, respectively.

The advanced design features improve the plant response to station blackout, loss of feedwater, and transient events, which are major contributors to core damage frequency in existing plants.

LEAD UNITS AT ULCHIN

Ulchin 3 and 4, currently under construction, are the lead KSNP units, and design work on two more KSNP units, Yonggwang 5 and 6, is scheduled to start next year. Without changing the skeleton of the KSNP design already developed, plant enhancements and improvements will be made as new design features not previously employed are proven to be safe and reliable (see Table 5). □



▲ Fig 8. Instrumentation and control system diagram for KSNP.

SCS piping and components will not be exceeded at this pressure, which reduces the possibility of an interfacing system loss of coolant accident if the SCS is exposed to the full reactor coolant system pressure.

The availability of the SCS during shutdown cooling operation will be increased by the elimination of the auto-closure interlock in the SCS suction valves. High reliability of the SCS during mid-loop operation is achieved by the installation of permanent refuelling water level indication and alarms.

The Containment Spray System (CSS) is a safety grade system and is designed to remove containment heat and reduce containment pressure following a main steam line break or loss of coolant accident. Fission products are removed by the spray so that, in the event of containment leakage, airborne activity at the site boundary will be reduced.

System reliability is increased by incorporating a dedicated containment spray heat exchanger in KSNP, whereas in the reference plant design the containment spray system uses the SCS heat exchanger. The interconnection between the SCS and CSS is nevertheless maintained in the KSNP so that, in the event of one or both of the shutdown cooling pumps being unavailable, one or both of the containment spray pumps can perform the SCS function (Fig 6).

Emergency Feedwater System. The Emergency Feedwater System (EFS) supplies feedwater to the steam generators to remove heat from the reactor coolant system to the SCS entry condition (350°F and 410lb/in²(a)) when the main and startup feedwater systems are unavailable. The EFS is configured into two separate independent subsystems (Fig 7). Each subsystem consists of one emergency feedwater storage tank, one 100% capacity motor-driven pump, one 100% capacity turbine-driven pump, and associated piping and valves. The design features include the capacity to supply feedwater for eight hours during station blackout.

The EFS has been simplified by the elimination of the cross-connection between trains and improvement of actuation logic. The system has been upgraded by making use of safety grade emergency feedwater tanks and venturis, prevention of common mode failure due to steam binding, and elimination of emergency feedwater isolation in the event of steam generator failure.

DESIGN FEATURES: CONTAINMENT

The containment provides a leak-tight barrier to preclude uncontrolled radioactivity release in the event of postulated accidents. The containment also provides protection for safety related systems and equipment from external hazards such as flooding or aircraft crash.

The containment building is a cylindrical structure with a hemispherical dome. It is constructed of prestressed concrete clad with a steel liner plate to endure thermal and dynamic loads arising from postulated accidents. Compared with previous designs, the reactor cavity has been enlarged and rearranged to capture and cool molten core debris should the reactor vessel fail under severe accident conditions.

Heat removal and fission product removal in the event of postulated accidents, such as main steam line break or loss of coolant, are performed by the Containment Spray System. All containment penetrations except those required to perform a safety related function during postulated accidents are isolated by the Containment Isolation System via the automatically initiated containment isolation signal.

Following a postulated loss of coolant accident, the Combustible Gas Control System prevents the uncontrolled recombination of combustible gas. A hydrogen igniter system has been added to control hydrogen concentrations following a severe accident, and to reduce the possibility of a hydrogen explosion by eliminating local high concentrations through controlled hydrogen burning.

DESIGN FEATURES: CONTROL ROOM

Modern technologies using software based control, automatic testing, multiplexing, alarm prioritisation, fault tolerance, and computer driven displays have been employed in the control room design (Fig 8). Human factor engineer-

Table 5. Summary of Korean PWR design concept development

	Existing plants	Standardized units	Future units
General			
● Capacity (MWe)	600 — 1000	1000	1000 or over
● Design life (y)	30 — 40	40	60
Safety improvement			
● Fuel thermal margin (%)	~5	5	15
● Core damage frequency	10^{-3} — 10^{-4}/reactor-y	10^{-4} — 10^{-5}/reactor-y	$<10^{-5}$/reactor-y
● Station blackout coping time (h)	Up to 4	4	>8
● Containment	Severe accident load not considered	Severe accident load considered	Severe accident load considered
● Source term	TID-14844	TID-14844 and realistic source term (PSA)	Realistic source term
● Hydrogen generation	Severe accident not considered	100% oxidation with 10 v/o limit	100% oxidation with 10 v/o limit
● Licensing	Severe accident not considered	Severe accident will be considered as a safety margin basis	Severe accident will be considered as a safety margin basis
Performance improvement			
● Plant availability(%)	~75	80–87	87
● Refuelling cycle (months)	12–15	12-24	24
● Load-follow	Limited load-follow	Planned load-follow	Daily load-follow with automatic frequency control
● Occupational radiation exposure (man-rem/y)	~350	100	<100
● Man-machine interface	Conventional analog control with some digital	Current advanced technology adopted with exceptions	Full advanced technology adopted

SYSTEM 80+™

Standard Design

THE Nuclear Option for the 90s...

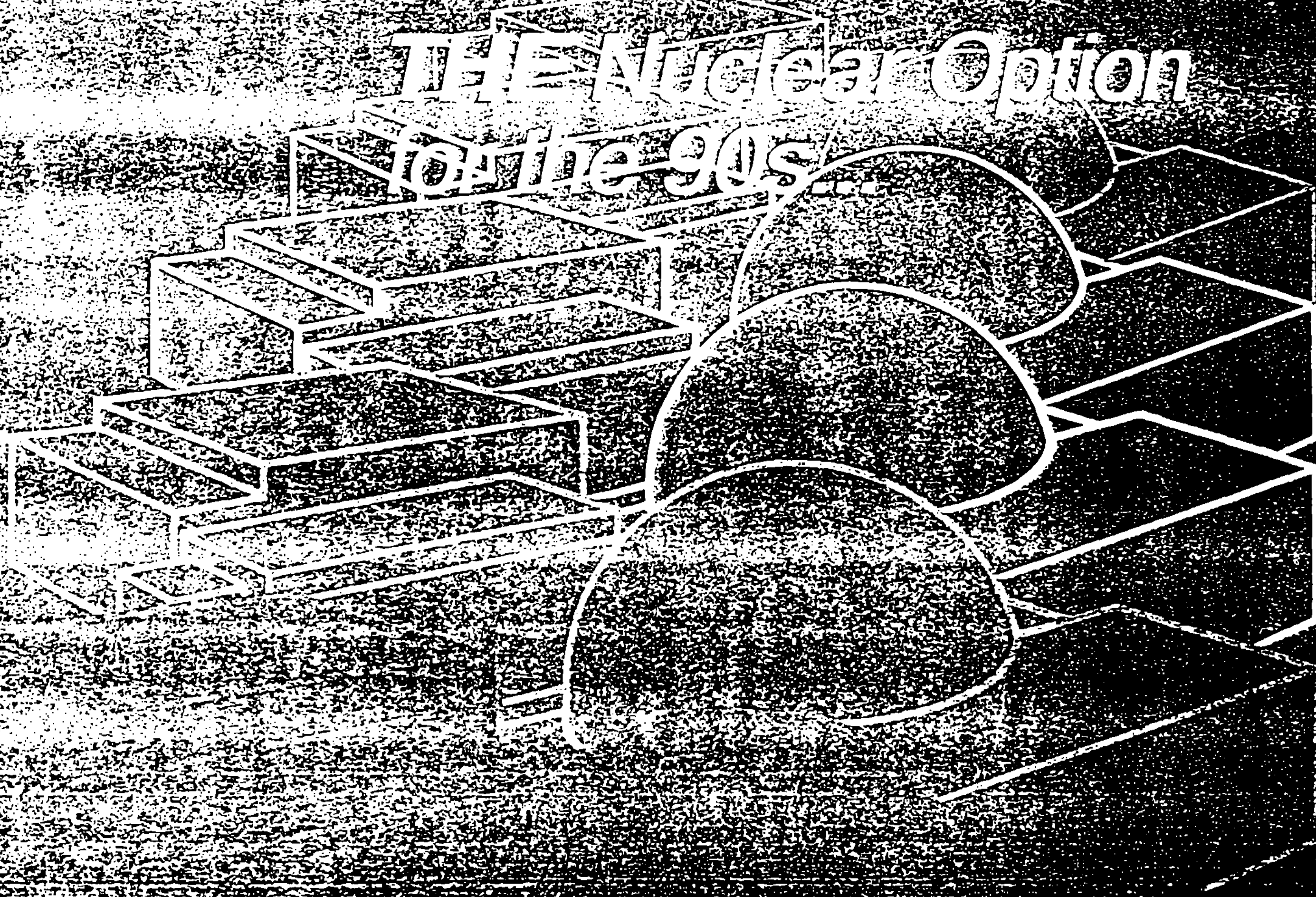


ABB Combustion Engineering Nuclear Power

ABB

ASEA BROWN BOVERI

The System 80+™ Standard Design

THE Nuclear Option for the 90's...

In 1991, nuclear power was second only to coal in use for generating electricity.

The demand for electricity is growing – it shows no signs of slowing down. To meet this demand new power plants will have to be built in the 1990's. Concerns about global warming, acid rain, nuclear safety, and the use of scarce resources, such as oil, to generate electricity all impact on the decision of how heavily to rely on any one energy source. Given the many options available - oil, natural gas, coal, nuclear, waste-to-energy, or even solar – there is no single correct answer. Rather, all feasible alternatives must be considered and, in the long run, each used where most practical.

Nuclear Power's Role

Today, there are 120 nuclear plants in operation around the country. More than 20% of all electricity produced in the U.S. comes from nuclear power, safely and without incident. Even so, we in the nuclear industry are well aware that the public's continuing concerns for safety and economic viability of nuclear power must be addressed and satisfied. That's why a new generation of nuclear power plants based on evolutionary design improvements and known as Advanced Light Water Reactors (ALWR) has been developed. Combined with the governmental reforms under way in the licensing of standardized nuclear plants, the nuclear industry is working hard to alleviate the public's concerns. In the following pages you will see how ABB-CE has developed an evolutionary ALWR design called the System 80+ Standard Design which is based on the most successful existing operating plants in the country. Factored into already successful designs are more safety systems, as well as improvements to make plant construction, operation and maintenance easier. When all factors are taken into account – environmental, economic, public safety, and energy security – ALWRs such as System 80+ must be considered a viable source of energy: one that deserves to be part of any plan for new capacity in the 90's.

Why the ABB-CE System 80+[tm] Standard Design?

The ABB Combustion Engineering (ABB-CE) System 80+ standard design evolved from a proven standard design, the original System 80[1] Nuclear Steam Supply System (NSSS) which is currently in operation and setting world records for power generation. Additionally, it meets the requirements of the Electric Power Research Institute (EPRI), a utility research organization that, with utility guidance, has formulated an industry-wide program to develop design requirements for the next generation of LWRs. It also measures up to the NRC's Standardization Rule (10 CFR 52) which defines technical requirements for future plants in the US. It's a design that has been submitted to the NRC for pre-licensing via design certification. In short, it's a standardized 1300 MWe design that's here now, ready for service in the 90's.

Evolving from Proven Concepts

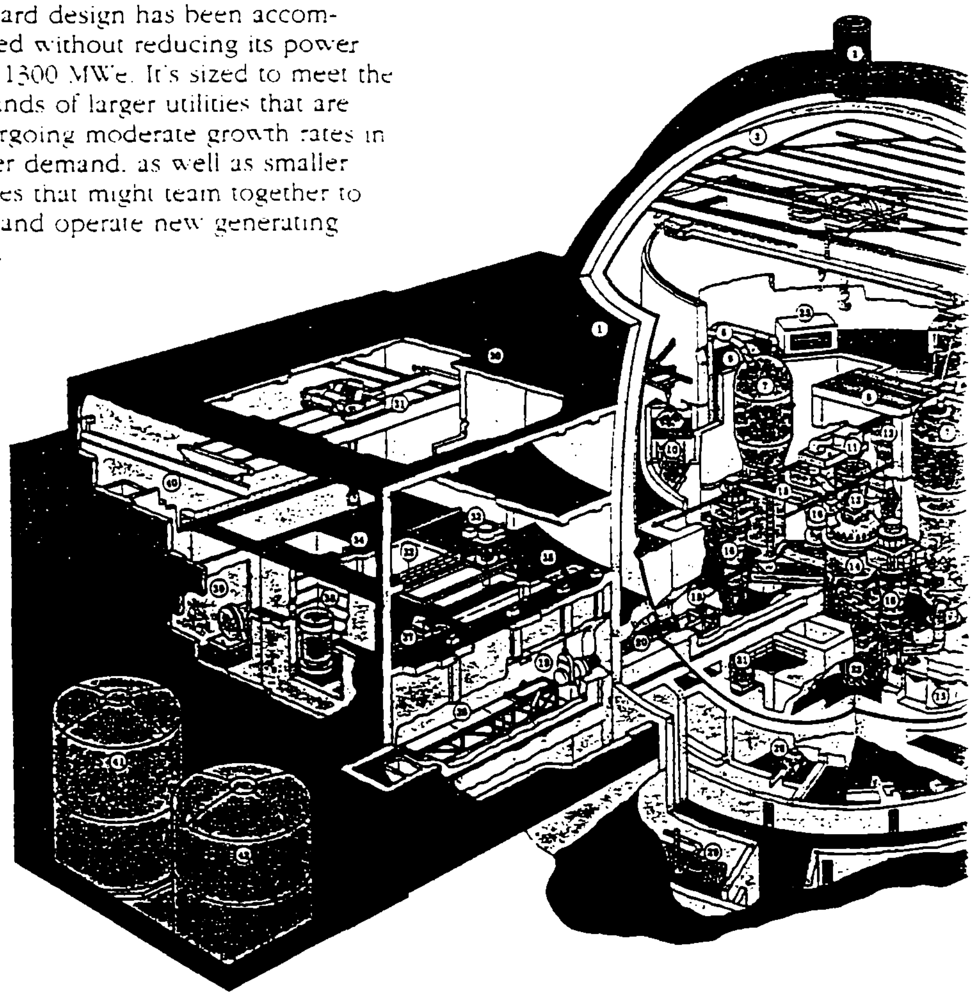
When the original System 80* standard design was first developed, it reflected the best that commercial nuclear power had to offer. That's why we've chosen to build on its success with the new System 80+ Standard Design. This evolutionary standardized design incorporates improvements which have evolved from thousands of reactor years of nuclear plant operating experience. Proven features include:

- digital mini computers for core monitoring and protection
- Nuplex 80+ human engineered control center
- long refueling cycles
- 100% load rejection capability

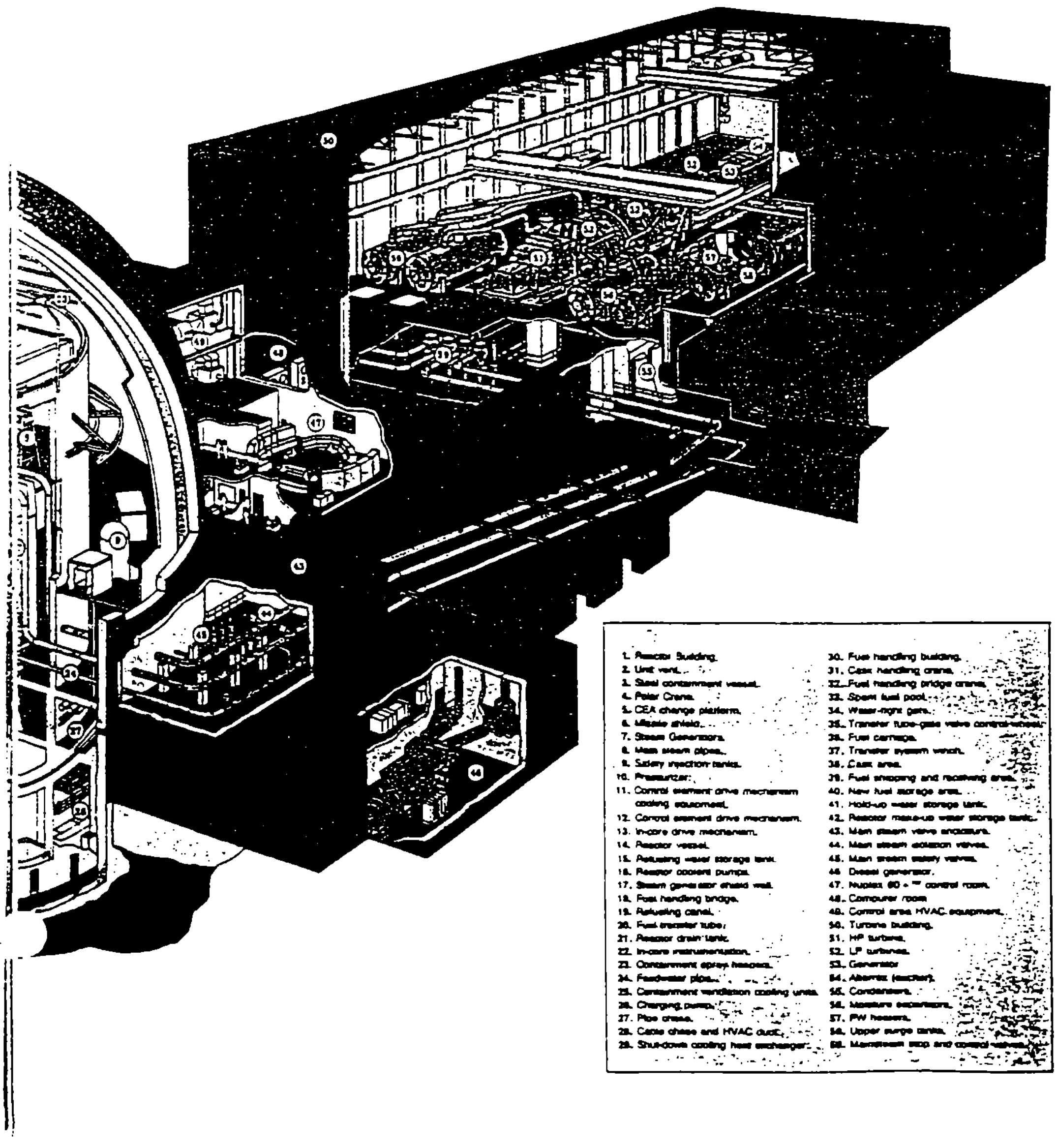
Utilities will recognize these features and many others as state-of-the-art on the System 80 design. At the Palo Verde plant in Arizona, unit three produced more total energy in one year (1988) – some 10.865.800 megawatt hours – than any other nuclear plant in the world. What's most important about this accomplishment is that, because of these units, System 80 represents the only US standard reference design available today with 100% of the details completed: with fabrication, construction and start up experience fed back into the design: and now, most importantly, with on-line operating experience to call upon. This experience at Palo Verde will be invaluable to the design and construction of future plants, as will the newest System 80 projects now underway with the Korea Electric Power Company in Yongwang and Ulchin, Korea.

Simplification with No Reduction in Power

Simplification of the System 80+ standard design has been accomplished without reducing its power from 1300 MWe. It's sized to meet the demands of larger utilities that are undergoing moderate growth rates in power demand, as well as smaller utilities that might team together to own and operate new generating units.



Palo Verde Nuclear Generating Station



Safe, Reliable Power

With safety and reliability as key watchwords, there is an important two step program in progress to develop and license the System 80+ Standard Design. First, the original System 80 design has been reevaluated and modified. Changes have been carefully made according to the ground rules set by EPRI and the nuclear utilities themselves in the industry-developed ALWR Requirements Document, as well as the requirements stated in the NRC's Severe Accident Policy.

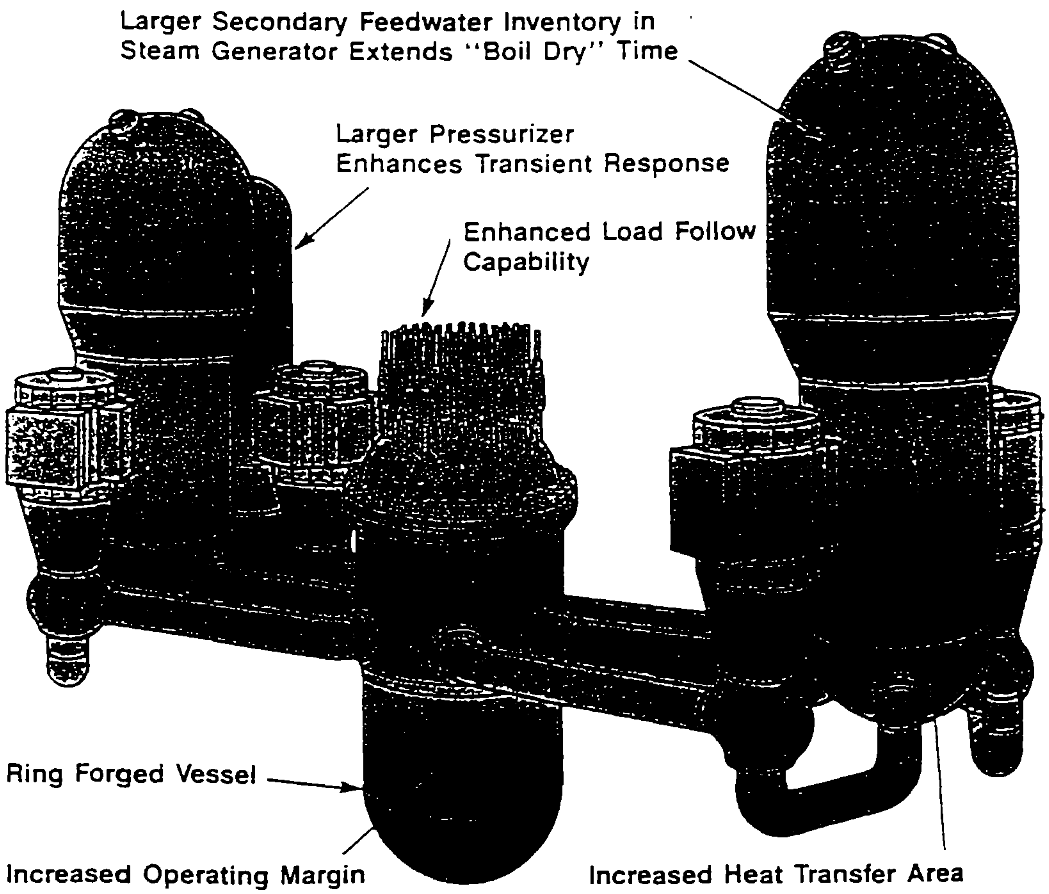
NRC certification is expected as early as 1993. Each step of the way, reliability and safety are being addressed. And each step of the way, ABB-CE has addressed these concerns by improving instrumentation and controls to simplify the coordination between man and machine; by creating redundancy in safety systems; and by developing an integrated planning approach for addressing maintenance, operation, construction, security and emergencies. The System 80+ Standard Design will satisfy the future needs of utilities, regulators and the public.

What new design features are in the ABB-CE System 80+[tm] Standard Design?

It's important to note that the evolutionary ALWR features in the System 80+ Standard Design are based on substantial improvements to the most successful of the existing nuclear plants. Safety, performance and economics have all been taken into account. Particularly noteworthy is that the evolutionary System 80+ standard design achieves safety levels that rival or exceed the smaller, more passive designs being developed. Yet System 80+ is ready to be built now – while the passive plants won't be ready for service until the late '90s. Although System 80+ will rely more heavily on operators than might passive plants, it will provide the operator with much more flexibility in handling unforeseen events than in the past. Literally hundreds of improvements have been made.

Rugged Dual Continment

System 80+ Standard Design improvements have been expanded to encompass the complete plant design. This includes the containment building as well. Basically, the containment structure is a steel sphere enclosed in a cylindrical concrete building. This provides the safety advantages of a dual containment by reducing the potential for radioactive releases to the environment and allows the potential addition of an external spray system to remove heat in the reactor in the event of a severe accident. And with a 200 foot diameter, its size gives it an added advantage in that any hydrogen that might be released during a severe accident can be safely diluted and controlled. Configuration of the reactor vessel cavity is such that it is able to cool molten core material and prevent it from being ejected directly into the containment atmosphere. Finally, the plant's safeguard systems are housed below the spherical containment structure and protected by thick concrete walls. This improves sabotage protection tremendously. A stronger containment building, specifically designed to withstand severe core damage accidents worse than occurred at Three Mile Island, and scores of other improvements, will provide renewed confidence in the safety and stability of nuclear power.



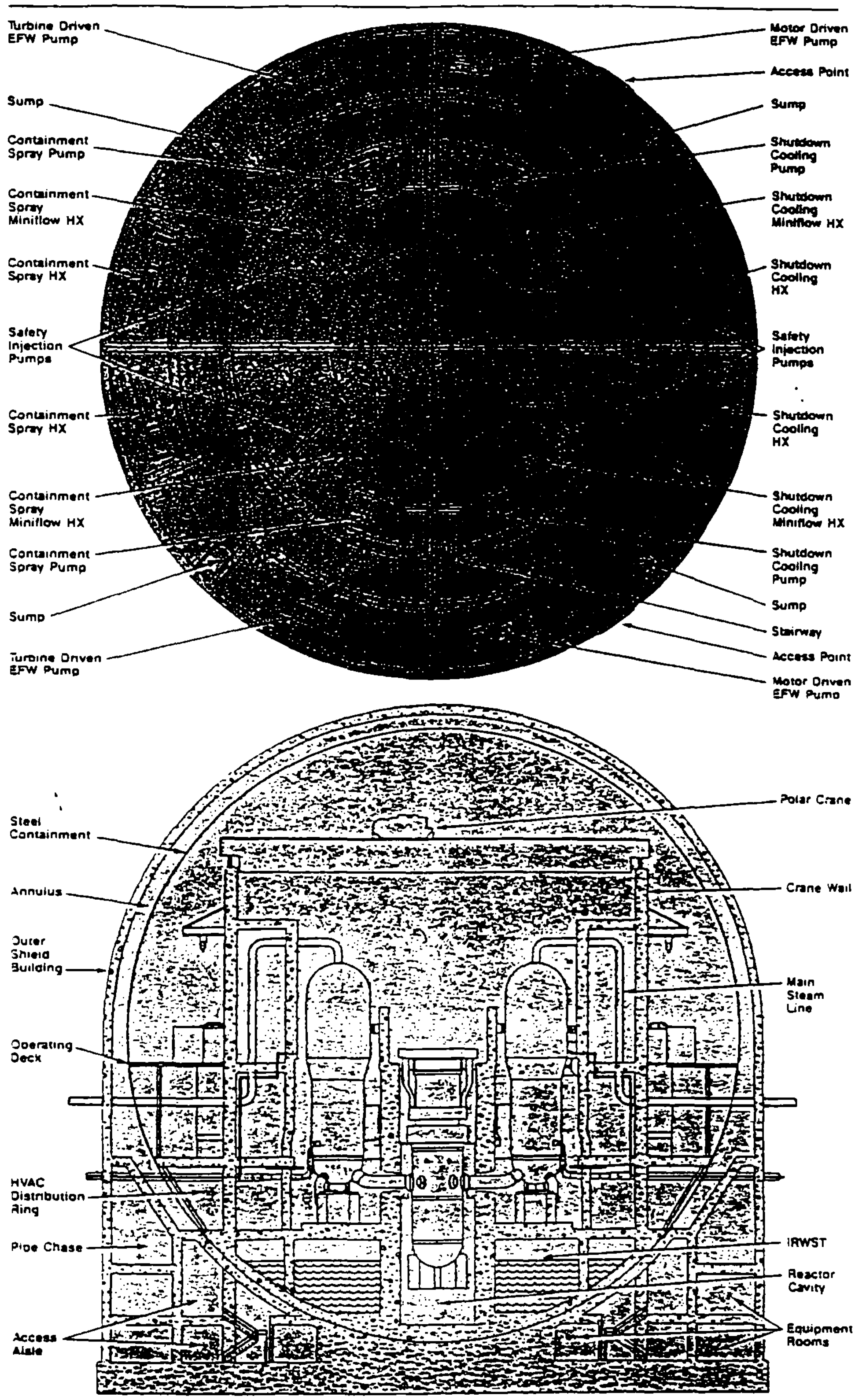
System 80+ Standard Plant - NSSS Enhancements

New Safeguards Systems

To reduce the adverse impacts of upsets in plant operation, ABB-CE has incorporated many new features which substantially increase the operating margin of the design. Improvements in the steam generator, for example, increase the water volume and heat transfer area to aid in cooling. More cooling capacity means longer cooling time and more time for the operator to act. Another newly designed feature (the Safety Depressurization System), provides a way to safely depressurize the reactor cooling system and provide protection in an emergency – even after a severe accident that damages the reactor. It also includes the capability to provide cooling in the event of a total loss of feedwater through a new "feed and bleed" process that avoids contaminating the containment building – feeding water into the reactor, bleeding steam out of the reactor and condensing the steam in a large tank of water at the bottom of the containment until a desired temperature is reached. The shutdown cooling system has been upgraded to a higher design pressure to make it easier for the operator to use and to prevent the chance that radioactive water could leak from the containment through this system.

Increased Redundancy

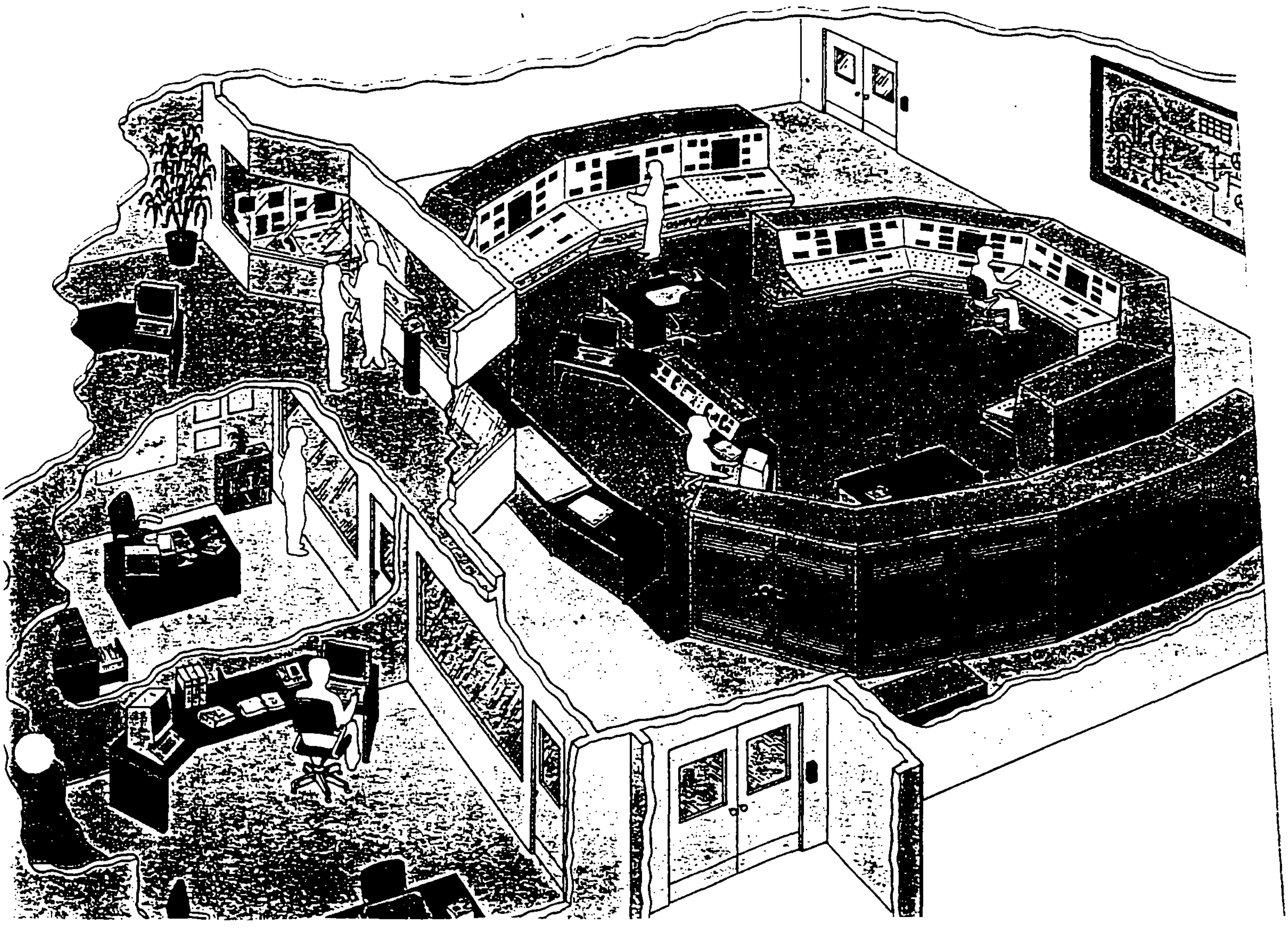
One of the most fundamental ways to increase safety is to create redundancy: multiple sets of systems that back up each other. The System 80+ Standard Design does just that with an array of features that improve not only safety, but reliability as well. Now there are four trains, or sources, of emergency feedwater to cool the reactor and protect it from overheating after being shutdown. To prevent any possibility of this feedwater being cut off, two sources are driven by electric motors, two by steam turbines. In still another effort to assure uninterrupted protection, pumps in the shutdown cooling system are identical to those in the containment spray system, and therefore, are interchangeable. As a result, complete water flow can be maintained in either system if a pump is not available. To guard against the loss of outside power to any part of the plant, another diverse power supply (a combustion turbine) has been added to the already existing diesel and battery back-up systems. In this way, the loss of any critical function is extremely unlikely. And finally, to minimize damage in the event of fire or sabotage, safety systems themselves are contained in separate quadrants of the plant. An event that damages one train of a safety system can't reach the others. With design redundancy in key plant areas, utilities are sure to find an extra margin of safety and reliability never before possible.

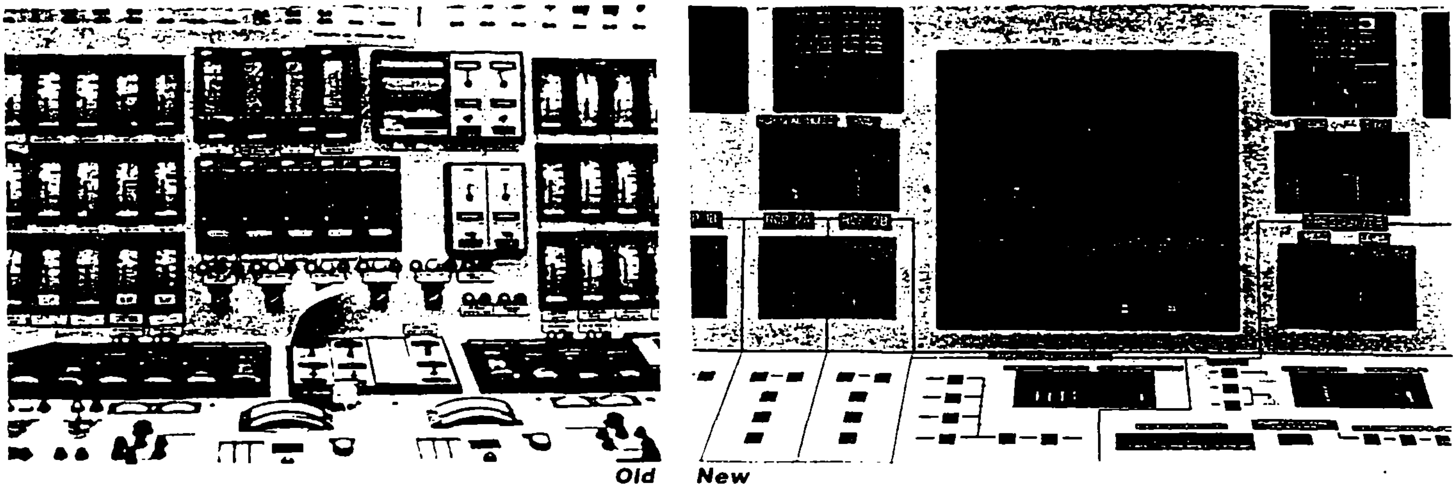


Spherical containment layout provides a large floor area with most of the space at the operating deck – where it can be used for maintenance and refueling activities.

What about the relationship between man and machine in the ABB-CE System 80+ Standard Design control room?

The TMI accident heightened the nuclear industry's awareness of the critical importance of the man-machine interface to producing safe, economical power. As a result, the most dramatic advances in the System 80+ design have been made where man ultimately faces machine – in our totally redesigned Nuplex 80+™ advanced control complex. What has happened in the last ten years might best be described as a revolution in the computer/electronics industry itself. Computers, mini-computers, programmable logic controllers and a host of other advances have all helped to dramatically improve the design of the control room and the way information is presented to the operator. For the System 80+ control room it represents a total revamp in the way information is collected, transmitted and processed for use by the operator – processed information that is directly usable by the operator for monitoring and control of the plant. The result is information that is easier to comprehend: simpler, but more complete. It's simpler because computers will reduce the amount of information continuously presented to the operator. Yet it's more complete because it will allow access to all plant data at the touch of a finger.





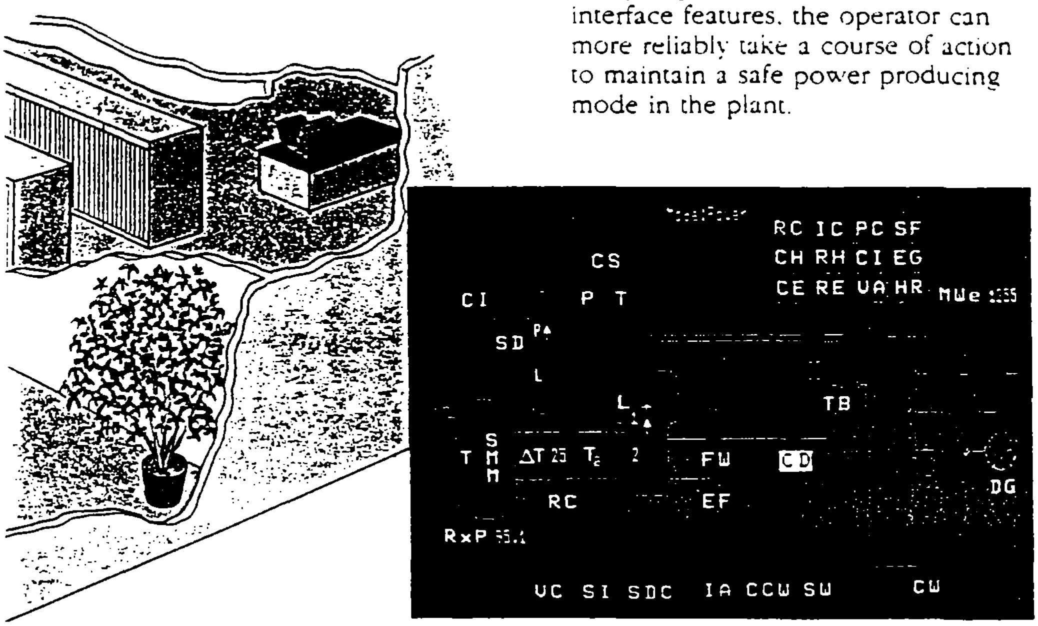
Nuplex 80+ significantly reduces operator information overload by using proven techniques.

The IPSO Display

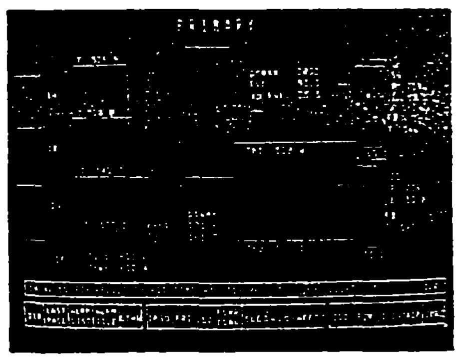
In what might be best described as the watchdog of the plant, a large, dynamic display screen in the control room provides an overview of the plant's status at all times. Named IPSO, for Integrated Process Status Overview, the display can also be accessed on touch-sensitive CRT units throughout the control room and in the adjacent Technical Support Center, as well as off site in an Emergency Operating Facility. From the control room panels, virtually all available information on the plant's status can be accessed. Progressing from the overview, more detailed information can be accessed on any particular system. Again, simply by touching a CRT screen, the operator can further "down-shift" to select information on individual components in that system and check for trouble. Preventing operator overload and minimizing confusion are fundamental to reducing human errors during both normal and abnormal situations. Through the use of such vastly improved man-machine interface features, the operator can more reliably take a course of action to maintain a safe power producing mode in the plant.

Unburdening the Operator

As an example, in earlier control rooms, there were as many as 19 different gauges to measure and indicate the varying ranges of pressure in the reactor coolant system. The operator was required to determine the most appropriate gauge for a given plant condition. Now the number of electronic pressure displays on the control panel is down to one, with the others available at the touch of a finger. With microprocessors analyzing and validating information to determine the most accurate readings, the operator is unburdened from the task of sorting redundant information: a critical factor in making him more effective and reliable in his job.



The "Big Board" Integrated Process Status Overview (IPSO) is visible throughout the control room to coordinate staff activities.



Carefully designed displays are used to provide information to the user.

Simplifed, Prioritized Information

Control room alarms have been improved in a number of ways. They have been prioritized to assist the operator in determining which actions to take and in what order. In addition, the number of alarms being displayed has been reduced considerably with lower priority alarms appearing on the CRT. As with all other signals, plant data is validated by computer before the alarms are activated. The result: simplified, prioritized information that is easier to understand and act upon.

U.S. NRC Issues Final Safety Evaluation Report on the System 80+™ Design

ON FEBRUARY 28TH THE NUCLEAR REGULATORY Commission (NRC) staff issued an advance copy of its Final Safety Evaluation Report (FSER) on the System 80+ standardized nuclear power plant design developed by ABB Combustion Engineering Nuclear Power together with Duke Engineering & Services Inc. and Stone and Webster Engineering Corporation. The System 80+ design is now set to receive Final Design Approval in August of this year.

Issued on-schedule and with all technical issues resolved, the FSER signals a major milestone in an immense licensing project and serves as proof that it is possible for government and industry to resolve safety issues while adhering to a schedule.

The FSER provides NRC staff acceptance of the System 80+ advanced plant design. Receiving Final Design Approval is essential for ABB's bid to construct Taiwan's fourth nuclear power plant. The Taiwan project is due to be awarded in the third quarter of this year and country of origin licensing is one of the major bid requirements.

The System 80+ plant is an evolutionary design based on the three System 80® units in operation at Palo Verde and the four units under construction in Korea. These Korean units incorporate a number of the advanced features of the System 80+ design. Continued implementation of System 80+ technology is under consideration for future nuclear plant development in Korea.

Design Certification is a process established in the U.S. NRC regulations which allows for the pre-licensing of standardized nuclear power plants in the U.S. The success of the project to license the System 80+ design is credited to the commitment shown by all of the involved parties to making the NRC's new Design Certification process work. Without this commitment, and the spirit of cooperation between ABB, its team members, utility advisors, the Department of Energy, and the NRC, no design, however good, could have hoped to navigate the new licensing process so quickly.

NRC's Dr. Thomas Murley, Director of Nuclear Reactor Regulation (left), congratulates Robert Newman, President of ABB CE Nuclear Systems.

Standardization and Design Certification

Plant by plant licensing meant higher costs	Design Certification of standardized designs saves time and money
Utility companies were committed to plant construction before designs were licensed and completed.	NRC approval is granted before utilities commit money to a project.
Nuclear vendors designed only a plant's nuclear components.	Nuclear vendors design the entire nuclear plant.
Utilities were responsible for designing the balance of the plant.	Nuclear vendors are responsible for pre-licensing their power plant designs before utilities commit to expensive construction projects.
Lack of standardized designs often meant lengthy construction and licensing delays.	Utilities are responsible for obtaining site licenses.
Delays meant that over half of a power plant's cost was due to interest charges.	Standardized power plant designs mean reduced costs.
Operating licenses were only granted based upon the analysis of completely constructed plants.	Standardized nuclear power plants are built in as little as half the time.
Utilities often had to make payments on completed but inoperative facilities while awaiting completion of the licensing process.	Pre-licensing eliminates delays.

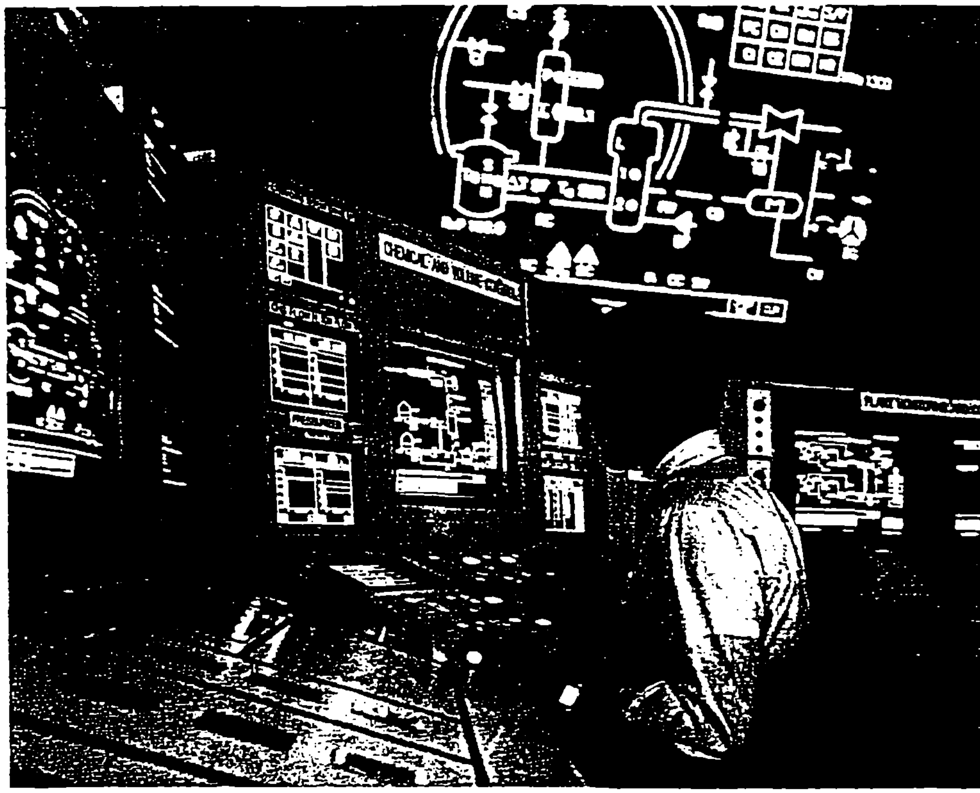
Issued on schedule and with all technical issues resolved, the advance FSER signals a major milestone in the nuclear industry.

ABB

The System 80+ Design Certification Schedule (per NRC staff schedule SECY-93-097)

1991		1992	
April 1989 — ABB-CE submitted the System 80+ design for NRC review	October — NRC issued the last official Request for Additional Information (RAI)	April — ABB completed responses to NRC RAIs	September — The NRC issued its Draft Safety Evaluation Report

An Advanced Control Room

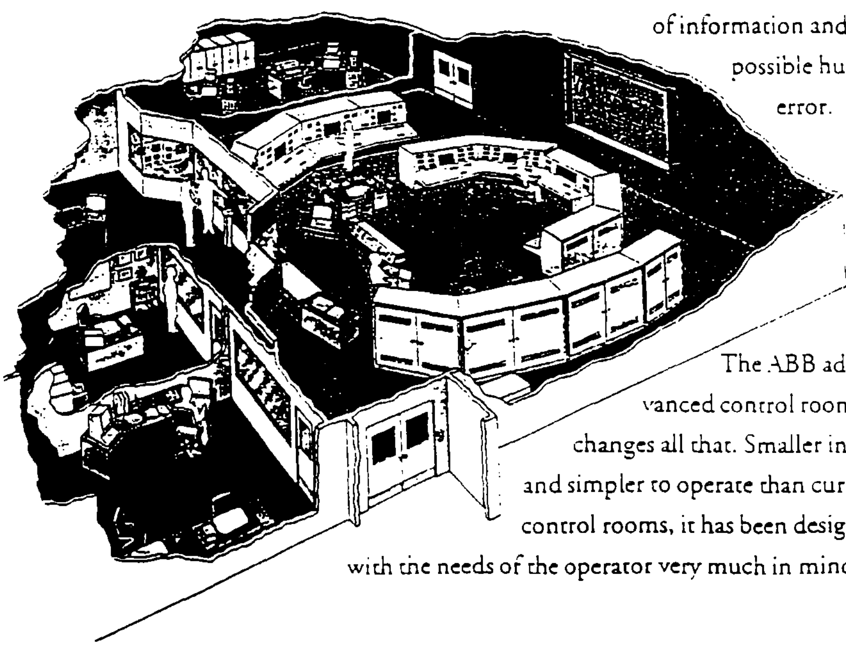


A new breed of control rooms.

IT USED TO BE THAT "USER-FRIENDLY" wasn't one of the first phrases one would use to describe the control room of a power plant.

Filled with massive amounts of knobs, switches, and indicators, conventional nuclear plant control rooms required from four to six operators. The multiple indicators and alarms sometimes result in an overload of information and possible human error.

The Nuplex 80+ Advanced Control Complex.



The ABB advanced control room changes all that. Smaller in size and simpler to operate than current control rooms, it has been designed with the needs of the operator very much in mind.

The Nuplex 80+ Advanced Control Complex uses digital protection, control and monitoring systems based on proven, commercially available hardware. Computers collect information about operating systems and interpret it for the operator. The readings from the sensors in the plant are verified and validated by the computers, and accurate readings are displayed on a control panel. When a problem occurs, computers pinpoint the malfunction and alert the operator, who can then easily access additional information on any system and take corrective action. The confusion has been eliminated.

So efficient is the control room, in fact, that it is able to be run by a single operator.

The Nuplex 80+ Advanced Control Complex has been an integral part of the review and approval process of the System 80+ design. In fact, it is now the only advanced control complex approved by the U.S. NRC.

1993 **1994** **1995**

DECEMBER
NRC approved the Inspections, Tests, Analysis, and Acceptance Criteria

JANUARY
ABB completed responses to all technical issues

FEBRUARY
NRC staff issued an advance copy of the System 80+ Final Safety Evaluation Report

APRIL
ABB will complete responses to all NRC comments and issues

MAY
Advisory Committee on Reactor Safeguards will document its review of System 80+ in a letter to the NRC

JUNE
NRC will formally publish the System 80+ Final Safety Evaluation Report

AUGUST
NRC will issue its Final Design Approval

SEPTEMBER
NRC will issue a Notice of Proposed Rulemaking to actively seek public participation in the certification process

DECEMBER
The NRC will certify System 80+ for construction and operation in the United States

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In what ways has the System 80+ design evolved from the System 80 design?

Major Enhancements from System 80 Operating Plants

Nuclear Steam Supply System

- Increased margins in reactor coolant system
- Improved steam generator design
- Four-train emergency core cooling and emergency feedwater systems
- Addition of a safety depressurization system
- Full flow testing capability in all safeguards systems
- Improved design of shutdown cooling and containment spray systems
- Nonsafety grade CVCS with centrifugal charging pumps
- Forged reactor vessel

Balance of Plant

- In-containment refueling water storage tank
- Use of auxiliary transformer to provide second source of offsite power
- Four safety bus arrangement
- Elimination of raw water within the nuclear island
- Spare feedwater train
- Addition of an alternate AC power supply

Instrumentation and Controls

- Human Factor Engineered advanced control room
- Big board overview display
- Alarm prioritization
- Mode dependence of alarms
- Data reduction and validation
- Sensor reduction
- Continuous on-line, automatic, self-testing
- Built-in expansion capacity of control panels and electronic equipment
- Modularity in design of equipment
- Extensive use of multiplexing and fiber optics

Radiation Protection

- Reduction of cobalt in primary system components
- Segregation of radioactive from non-radioactive systems
- Provisions for shielded maintenance access and workspaces
- Flushing and/or decontamination capability for primary and radwaste systems

Structures / Architecture

- Dual, spherical containment
- Entire nuclear island on common base mat
- Structural boundaries provide separation of safety equipment
- Dedicated aisles and corridors for operations and maintenance access
- Abundant laydown area near all equipment

Working as a Team Produces an Integrated Design

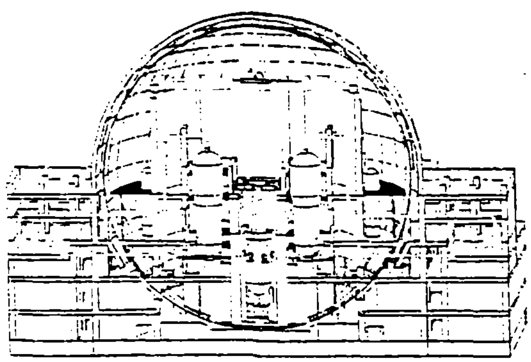
THERE AREN'T TOO MANY PROJECTS which require as much precise coordination as designing a power plant.

That's why ABB put together a single integrated team to develop the System 80+ advanced nuclear power plant. Having everyone on the same team and devoted to a single project ensured that all of the different elements in the design would fit together seamlessly.

When the Nuclear Regulatory Commission (NRC) licensing review process began, the sheer number of design details required that they be handled in as efficient a manner as possible. Issues raised by the NRC were divided into twenty "chapters," the responsibility for resolving the issues in each chapter falling to a single "chapter champion."

Delegating responsibility in this manner was necessary in order to meet important deadlines. While each chapter champion dealt directly with his or her counterpart at the NRC, meetings between chapter champions were held weeklyso as to insure that design changes were well coordinated.

Because at ABB, system integration isn't an abstract concept; it's a daily event.



Big on Safety

When it comes to safety, there's little point in thinking small.

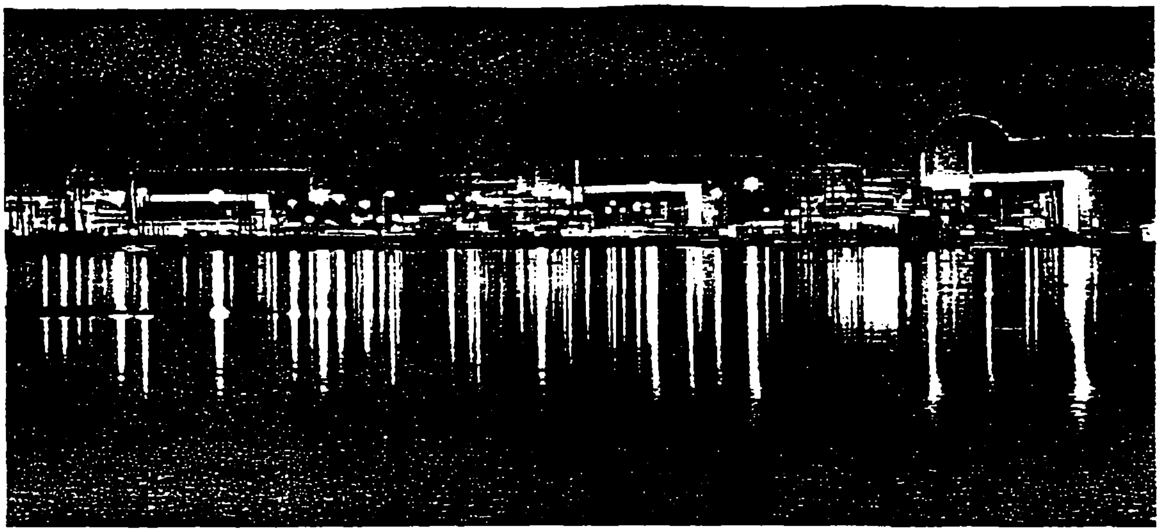
That's why the System 80+ Final Safety Analysis Report comes in 26 volumes - hardcover. Submitted to the NRC for approval, this report documents all facets of the System 80+ design related to safety.

But while those 26 volumes may tell the whole story, they don't tell how that story was written. The NRC subjected the System 80+ design to a level of scrutiny heretofore unprecedented: during the licensing review, ABB answered more than 4,500 detailed questions from the NRC.

Together, ABB and the NRC have spent well over 50 million dollars and one million hours on the licensing review of the System 80+ design.

At ABB, when it comes to safety, we like to think BIG.

The System 80+ nuclear island, featuring spherical containment.



The three standardized System 80 units at Palo Verde Nuclear Generating Station in Arizona.

Evolutionary Designs: Building upon Excellence

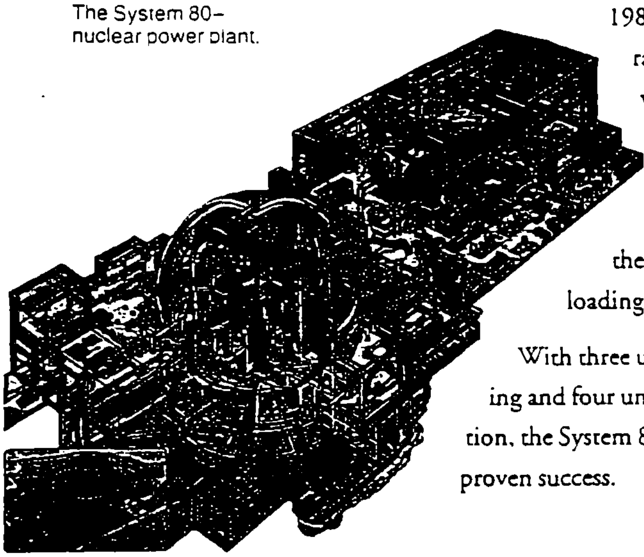
AT ABB, EVOLUTION MEANS STARTING WITH THE BEST and improving upon it.

The evolutionary forerunner to the System 80+ plant design is the System 80® Nuclear Steam Supply System. The System 80 design has demonstrated its soundness through superior performance and an exemplary safety record.

The System 80 design has demonstrated its soundness through superior performance and an exemplary safety record at Palo Verde.

Three System 80 nuclear units make the Palo Verde plant in Arizona the country's largest nuclear power facility. In addition to being the first standardized nuclear reactors built in the U.S., these three units have set numerous safety and performance records (see box).

Four more System 80 plants are presently under construction in the Republic of Korea. ABB has been involved in a comprehensive technology transfer program with Korean industry since 1987. Already incorporating many advanced features taken from the System 80+ design, the first of these four units will be loading fuel this year.



The System 80+ nuclear power plant.

With three units already operating and four units nearing completion, the System 80 design is a proven success.

That's why, when developing the System 80+ nuclear power plant, we started with the System 80 design.

When you've got something this good, there's no reason to start over.

System 80® Plants Set Performance Records

- **1988** Palo Verde generates more electricity than any other U.S. nuclear facility.

  Unit 3 has the fastest start-up testing and longest initial run in the U.S.

  Unit 3 generates more electricity in a single year than any other unit in the world.
- **1990** Palo Verde generates more electricity than any other U.S. nuclear facility.
- **1991** Palo Verde generates more electricity than any other U.S. power facility—nuclear or fossil.
- **1992** Palo Verde generates more electricity than any other U.S. nuclear facility.
- **1993** Unit 3 generates more electricity than any other U.S. nuclear unit.

ABB

ABB-CE's System 80+™ Standard Design. Easier to operate, easier to maintain.

Two of the fundamental objectives underlying the improved design of System 80+ are simple in themselves: Make it easier to operate and make it easier to maintain. Great care has been taken to be certain the operator's job is easier; that there is more time to respond and make decisions; and that the decision-making process itself is simpler and easier thanks to the Nuplex 80+ control room. At the same time, maintenance has also been made easier, with easier access to critical components; simpler designs that are easier to maintain and scores of other improvements geared to increase plant reliability and reduce the likelihood of operator or maintenance error.

Separating Safety and Non-Safety Functions

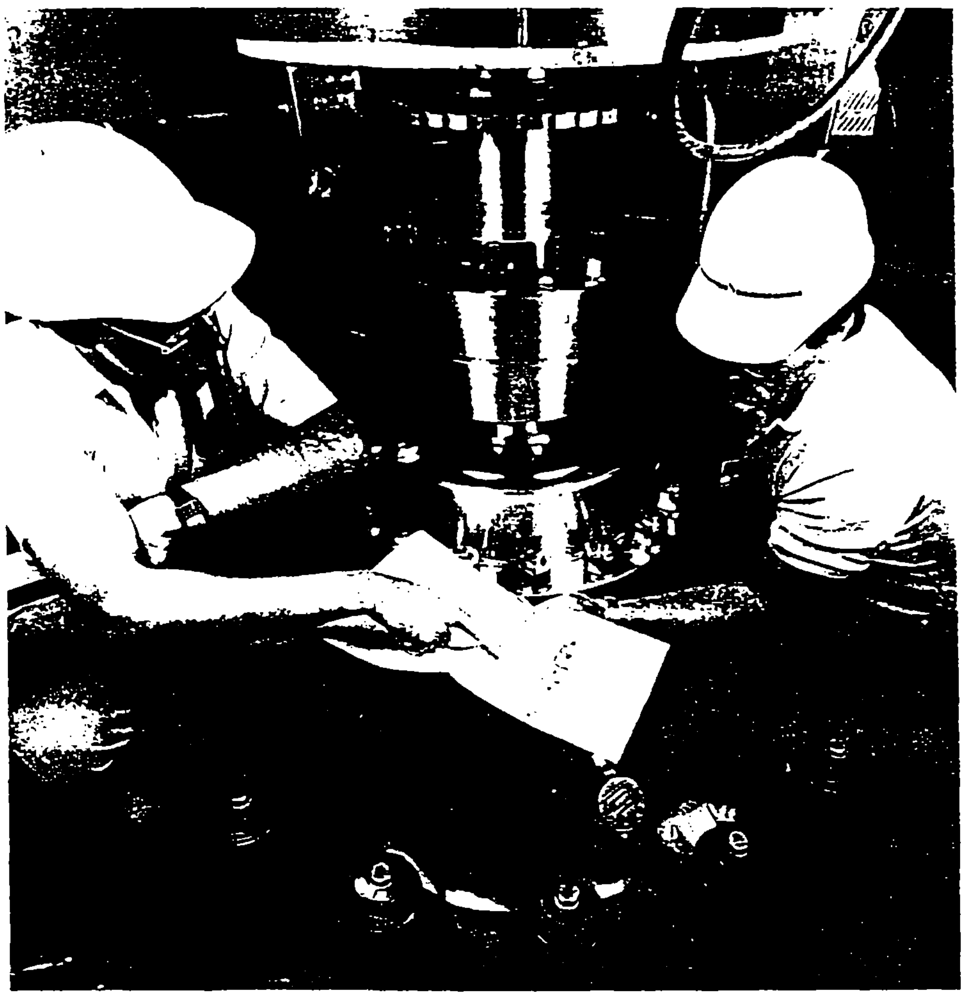
In existing plants many systems are called upon to perform more than one function. Very often, this means performing a safety function and non-safety function with the same piece of equipment. In almost all cases involving the System 80+ Standard Design, safety and nonsafety functions have been separated, making life easier for the plant operator and simplifying the maintenance process as well.

More Room for Maintenance

The steel spherical containment, in addition to other advantages cited earlier, offers another big plus when it comes to maintenance – more room. At 200 feet in diameter, it will be among the largest containment structures in the world. Because of the spherical shape, the operating deck will be much larger than those found in any current US plant. This will mean more space for performing maintenance activities during scheduled plant outages.

Self Testing

In designing the System 80+ control room, many improvements were made. One significant change is the inclusion of self testing features into many of the plant systems by using the latest in continuous on-line, automatic computer aided testing



circuitry. Although it sounds complex, the end result is further simplification. Personnel errors, usually associated with maintenance and testing on plant systems, will be significantly reduced as a result of this procedure. Overall, it will not only simplify activities of plant personnel but it will also reduce the number of unplanned reactor shutdowns due to operator error.

VPBÉR-600 NEW-GENERATION ENHANCED-SAFETY POWER REACTOR

F. M. Mitenkov, G. M. Antonovskii, V. S. Kuul', Yu. K. Panov, O. B. Samoilov, L. N. Frelov, N. N. Ponomarev-Stepnoi, N. E. Kukharkin, and Yu. G. Nikoporets

UDC 621.039.58

The present stage of development of nuclear power engineering both here and abroad is characterized by the high-priority task of increasing the safety of operating nuclear power plants and the creation of reactors of enhanced safety for nuclear power stations of the new generation. The outlook for the development of nuclear power is uniquely determined by the possibility of guaranteed safety of the population and environment. A high level of safety is achieved by the improvement of active and introduction of passive protective and containment systems as well as the successive realization of the concept of inherent safety. The creation of new-generation reactors with self-protection will ensure immunity to equipment malfunction and human error, limit the radiation effects of the most serious accidents, and make it unnecessary to evacuate the population.

Considerable simplification of the systems by the use of passive safety systems, economical fuel cycles, modular and high-quality fabrication, and proven and high-longevity equipment will make it possible to improve the economic factors of new-generation nuclear plants with reactors of enhanced safety.

The problem of creation of a reactor with the highest attainable safety has been solved successfully as applied to the reactor apparatus of the AST-500 nuclear heating station, whose high safety has been confirmed by the experts of the IAEA [1, 2]. Therefore, the fundamental safety features of the AST-500, such as an integrated reactor construction, the use of a containment vessel, and passive safety systems with different operating principles with large margins and self-actuation have provided the basis for the 630-MW VPBÉR-600 enhanced-safety power reactor.

Reactor Construction. The VPBÉR-600 is a water-moderated–water-cooled integrated reactor in a containment vessel for localization of accidents associated with depressurization of the pipes of the first loop or vessel. The integrated implementation is characterized by the placement in a single vessel of the core with working organs of the core-control system (CCS) and the heat-exchange surface of the steam generator and pressure compensator, whose function is performed by the upper space of the reactor vessel above the coolant level (Figs. 1 and 2). Heat removal from the core is accomplished by forced coolant circulation by means of the main circulation pumps of the first loop, which are built into the bottom of the reactor vessel. The vertical centrifugal pump has a hermetically sealed electric motor.

Above the core is a unit of tubes and devices that seal the fuel-element-section caps that contains guides for the CCS working organs. Coolant is supplied uniformly to the tubular steam-generator systems through openings in the intravessel well. In an annular gap between the reactor vessel and the intravessel well at a level above the core is the heat-exchange surface of the once-through steam generator, which consists of independent sections. At the entrance to the core is a pressure chamber, which ensures a uniform distribution of coolant flow over the fuel-element sections. Owing to the simplicity of the loop, a high level of natural circulation is created, which is sufficient for reliable core cooling in all emergency situations, including steam condensation when the water circulation is interrupted.

A feature of the integrated arrangement is the presence of a large water space between the core and the reactor vessel, which acts as a radiation shield. This reduces the neutron fluence to such an extent (to $5 \cdot 10^{16}$ cm^{-2}) that changes in the properties of the metal of the reactor vessel under the influence of radiation are not a problem. Above the steam generator and below the coolant level are heat exchanger–condensers for emergency removal of heat from the reactor, which operate as condensers when the first loop is depressurized. On the reactor cover are individual electromechanical CCS drive mechanisms, by means of which their working organs are moved for control and are dropped into the core in case of emergency. Ionization chambers are placed in the space between the intravessel well and the reactor vessel at the core level.

Special Design Office of Industrial Engineering. Russian Scientific Center "Kurchatov Institute." Translated from Atomnaya Énergiya, Vol. 73, No. 1, pp. 6-13, July, 1992.

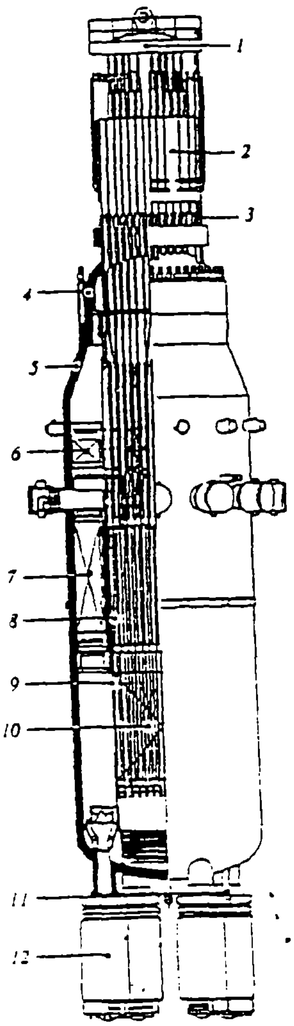


Fig. 1

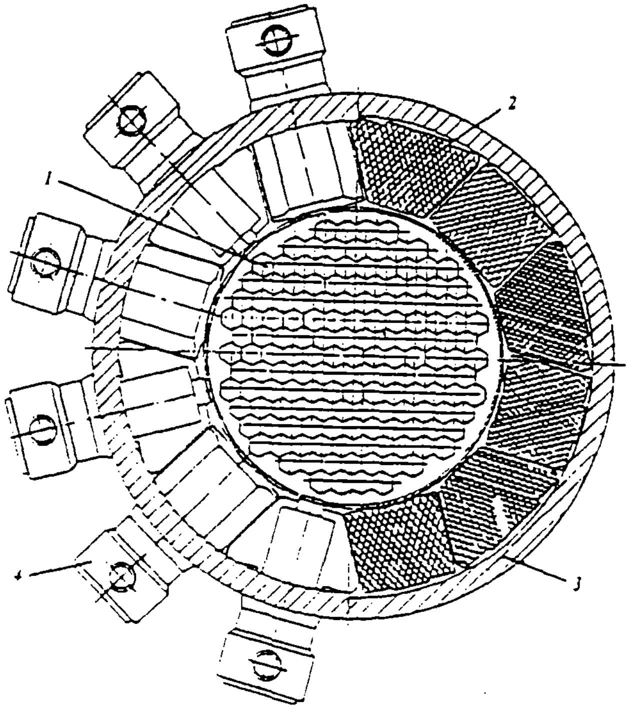


Fig. 2

Fig. 1. Overall view of VPBÉR-600: 1) upper unit; 2) drive mechanism of core-control system; 3) level meter; 4, 5) reactor cover and vessel; 6) heat exchanger–condenser; 7) steam generator; 8) unit that seals fuel-element-section caps; 9) intravessel well; 10) core; 11) ionization chamber; 12) circulation pumps of first loop.

Fig. 2. Cross-section of VPBÉR-600: 1) fuel-element section; 2) steam-generator section; 3) reactor vessel; 4) pipe of steam-generator section.

The core of the VPBÉR-600 consists of 151 fuel-element sections with fuel elements similar to those used in the VVÉR-1000. Each fuel-element section contains absorbing elements — rods with cadmium boride, which are combined in a cluster and form the working organ of the core-control system. In the core of the VPBÉR-600, the parameters of the fuel lattice are the same as in the VVÉR-1000. The boric-acid content of the coolant is reduced as compared with the VVÉR-1000 to ensure for the adopted fuel-lattice parameters negative values of the steam coefficient of reactivity and the coefficient of reactivity with respect to coolant temperature over the entire range of operating temperatures. Negative feedback for coolant temperature ensures power self-limiting under emergency conditions with an increase in power and temperature in the core; a negative steam coefficient of reactivity ensures self-shutdown of the reactor with depressurization of the first loop. A highly effective mechanical system is used to compensate for reactivity in the core. Most of the fuel-element sections of the core have CCS working organs. The mechanical system makes it possible to achieve reactor subcriticality in the cold depoisoned state in the absence of boric acid in the moderator with allowance for suspension of the most-effective organ in the extreme upper position.

The reactivity margin for fuel burn-up is compensated for jointly by the CCS mechanical working organs, the boric acid in the coolant, and the self-shielding burn-up absorber. The use of gadolinium in a mixture with the fuel is being considered as an absorber.

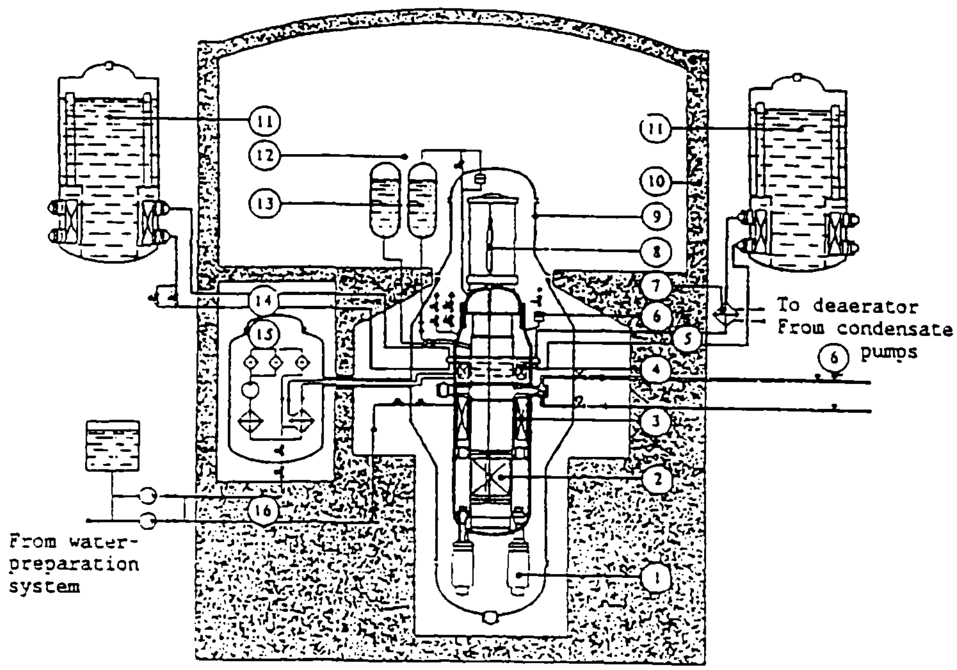


Fig. 3. VPBÉR-600 reactor: 1) main circulation pump: 2) reactor: 3) steam generator: 4) heat exchanger–condenser: 5) system for continuous heat removal: 6) self-actuated direct-action device: 7) intermediate heat exchanger: 8) CCS drive mechanism: 9) containment vessel; 10) protective envelope; 11) heat-exchanger unit: 12) system for emergency boron introduction: 13) container with boron solution: 14) system for passive heat removal: 15) system for purification and boron compensation for reactivity: 16) make-up system.

The average core energy is 69.4 kW/liter, and the average linear power of the fuel elements is 114 W/cm. The moderate energy intensity makes it possible to achieve high heat-engineering margins in normal operation and planned emergencies as well as safety of the installation in unplanned emergencies. To increase fuel burn-up, provisions are made for rearrangement of the fuel-element sections with placement of the most burned-up fuel mainly at the periphery of the core, which reduces neutron leakage.

The core allows the realization of fuel cycles with different recharging multiplicities and operating intervals between recharging. The main characteristics of one of the core-recharging regimes are as follows:

Core equivalent diameter, cm	305
Core height, cm	353
Number of fuel-element sections	151
Average enrichment of make-up fuel, %	4.15
Number of rechargings per run	4
Interval between reloadings, years	1.5
Average burn-up fraction of unloaded fuel, MW·day/kg	50

The reactor and all of the systems that are under the pressure of the first loop during operation and placed in a containment vessel. Location of the reactor in a second strong vessel designed for the pressure that arises with depressurization of the first loop ensures that the core remains below the water level, prevents fuel melting, and serves as an additional passive barrier for containment of radioactive products. The reactor is located inside a strong and dense concrete protective envelope.

The development of the VPBÉR-600 is based on experience gained in the creation and successful operation of nuclear-powered icebreakers and AST-500 heating plants. The engineering solutions regarding equipment such as the leak-proof main circulation pumps, the once-through steam generator, the containment vessel, and the passive self-actuating devices have been reliably assimilated and mastered. The leak-proof main circulation pumps, whose loading is not less than that of the pumps of the VPBÉR-600, have been thoroughly developed and tested in nuclear-powered icebreakers for more than 20 years with more than 100,000 hours of operation without loss of performance. The technology of pump fabrication has been mastered, especially in mass production. The design of the once-through steam generators has been checked experimentally, and the steam generators have undergone comprehensive tests and operated with the required characteristics in nuclear plants. The technology of fabrication of steam generators with a large number of steam-generating elements has been mastered and the mass production of such elements has been organized.

A containment vessel of similar construction is used in the AST-500 and has been thoroughly checked and in substantiation of the design and acceptance tests. Compartments for transport nuclear power plants whose size and materials corresponds to those of the containment vessel of the VPBÉR-600 are being manufactured. Passive self-actuated devices such as pressure-operated circuit breakers, hydraulically controlled pneumatic distributors, and membrane fuses, which are widely used in the AST-500 and nuclear-powered icebreakers, have been tested and accepted for production. The principal technological problems associated with reactor-vessel fabrication have been solved. The design of the VPBÉR-600 equipment permits thorough bench testing of individual elements, sections, units, etc. with confirmation of all specifications before being put into operation at a nuclear power station.

Layout of Reactor. The VPBÉR-600 is a two-loop water-moderated–water-cooled integrated-vessel reactor (Fig. 3). The first loop includes the main circulation loop, which is located inside the reactor vessel, as well as systems for pressure compensation, coolant purification, and liquid-absorber extraction. This loop is serviced by systems for water preparation, filling, and make-up, sample taking, air removal, and drainage. The second loop consists of 12 independent steam-generator sections with independent water supply and steam removal beyond the containment vessel. The steam-generator sections are joined in four loops, through which the steam is delivered to the turbine, from which the supply water is returned. The heat-engineering parameters are maintained by means of the flow rate of supply water through the steam generator and control of the first-loop temperature.

The accepted statistical parameters are close to the natural characteristics that correspond to self-regulation and ensure the pressure in the reactor changes insignificantly with various load combinations. Normal cooling shut-down is accomplished by the circulation of supply water through the steam generator with steam ejection into a special technological condenser. The principal characteristics of the VPBÉR-600 are as follows:

Heat power, MW	1800
Electric power, MW	630
First-loop coolant circulation	forced
First-loop coolant parameters:	
pressure, MPa	15.7
temperature at core entrance, °C	294.5
temperature at core exit, °C	325
Second-loop parameters:	
steam output, tons/h	3350
superheated-steam pressure, MPa	6.5
superheated-steam temperature, °C	305
Power-variation range, $\%N_{nom}$	30-100
Operating life, years	60
Maximum earthquake rating on MSK-64 scale	8

Safety Systems. The reactor scram system is designed to stop, slow down, or limit the chain reaction in an emergency situation or with deviation from normal operating conditions. This is achieved by sending the corresponding instructions to the core-control systems and by subsequent insertion into the core or prevention of extraction of the working organs. The highest level of protection provides for the insertion at maximum speed of all CCS working organs (dropping by deenergizing the drive motors). The protection is actuated with emergency levels of reactor power or doubling time, overpressurization or depressurization of the first loop, an earthquake, the loss of station electrical power, cessation of supply-water delivery, or pressing of the SCRAM button in the main or spare control panel.

An important feature of the VPBÉR-600 is the use of self-actuating devices that deenergize a part of the CCS drives that is sufficient for functioning of the safety system, bypassing the automatic circuit for the direct action of pressure in the reactor or containment vessel. A system for emergency boron introduction brings the core to a subcritical state and holds it there when

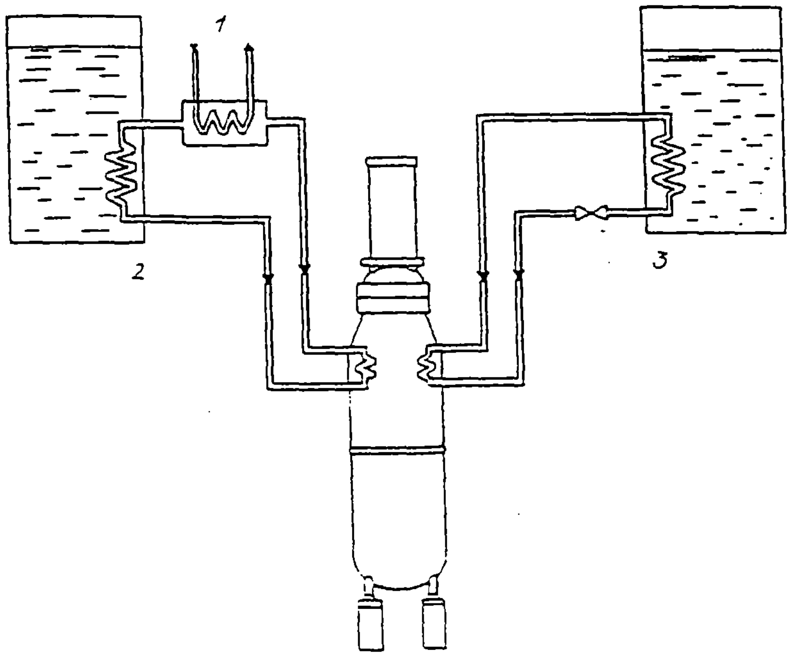


Fig. 4. System for emergency heat removal: 1) cooling water; 2, 3) systems for continuous and passive heat removal, respectively.

the reactor is in a cold or depoisoned state in case of nonactuation of the CCS drives and consists of two containers with boron solution. One of the containers is activated by opening a pneumatic fitting in the pipes connecting the container with boron solution to the reactor or by rupture of the membrane by the direct action of pressure in the reactor. The boron solution is fed by gravity into the core after equalization of the pressure in the reactor and the container by placement of the latter above the reactor. The second container, which is designed for reactor shut-down in emergencies associated with depressurization of the first loop, is passively activated by the hydraulic-accumulator principle.

The passive systems for emergency heat removal include two heat-exchanger units, which form systems for continuous and passive heat removal, each of which transfers the heat to reserve-water tanks. The systems for continuous and passive heat removal operate by natural coolant circulation (Fig. 4). The heat is extracted through an intermediate loop at a pressure higher than that in the reactor. The system for continuous heat removal functions constantly. The heat removed by the system from the reactor in normal operation heats the supply water of the steam generators in the intermediate heat exchangers. The system for passive heat removal is activated in an emergency by the opening of valves in the pipes that drain the water from the water heat exchangers on signals from the automatic-control system or directly from the action of pressure or the water level in the reactor.

With shut-down cooling, the heat removed from the reactor by the heat exchanger–condensers of the systems for passive and continuous heat removal is transferred to the reserve-water tanks, through which the process water circulates. In case of cessation of process-water circulation, the heat is removed by the evaporation of water from the tanks. The water reserve in one tank is sufficient for shut-down cooling over a period of 3 days; the water in the two tanks, a period of 7 days. Air heat exchangers can be used for a limited time to remove heat from the reactor after the water has been evaporated from the tanks.

The first loop is protected from overpressurization without the ejection of radioactive coolant by the reliable removal of heat from the core by the systems for continuous and passive heat removal. However, considering the fast dynamics that is characteristic of installations with forced coolant circulation, to limit the running out of pressure, relief valves are provided that eject the vapor–gas mixture from the pressure compensator of the reactor into the boron container or containment vessel without the escape of radioactive coolant from the reactor.

The reactor's decompression system is designed for an unplanned emergency with depressurization in its lower part. With an increase in the pressure in the containment vessel and a considerable reduction of the coolant level in the reactor, the direct effect of these parameters opens the decompression valves in the pressure compensator, which equalizes the pressures in the reactor and containment vessel and stops leakage.

Localizing Safety Systems. The containment vessel is a passive protective and localization device that ensures safety when pipes are ruptured or the reactor vessel is depressurized within the limits of the technically possible value. It is designed for the pressure produced by depressurization of the first loop and serves as a means for keeping the core below the water level, shut-down cooling of the reactor, and containment of radioactive products.

The localization system switches off the steam generator in case of failure of the seal of its heat-exchange surface or pipes. Each section of the steam generator has an electrically driven localization system, which is welded to the containment vessel. Each loop has two pneumatically driven localization systems, one of which is actuated by signals from automatic-control systems while the other is activated by automatic-control systems and a direct signal indicating level reduction in the reactor, and a check valve and a pneumatically driven localization device are mounted on the supply-water pipes.

Safety. The safety concept of the VPBÉR-600 is a combination of internal self-protection and passive safety systems. The physical properties of the core and the characteristics of the pressure compensator allow shut-down or power limiting without the use of the scram system, owing to strong negative power–boiling feedback. The system for continuous heat removal is constantly in operation and therefore does not require actuation of any devices when emergency shut-down cooling is required; the similarly implemented self-actuated passive heat-removal system guarantees highly reliable emergency heat removal. The first-loop coolant of the VPBÉR-600 greatly increases the total heat capacity of the system, which is capable of storing the residual heat release of the core, which provides large time reserves for the realization of emergency control measures. The integrated implementation principally eliminates the accident classes of large and moderate leakage when the first-loop pipes are ruptured.

Placement of the reactor in a strong containment vessel ensures that the core is kept under water with any rupture of the first loop, which prevents fuel melting. The containment vessel is an additional passive barrier for localization of radioactive products.

An attractive property of the VPBÉR-600 is self-realization of a safe state. The inertia of the reactor and the large amount of time available in severe emergencies leave no doubt that personnel will be able to control accidents and serve as a guarantee of safety. The probability of an accident with severe damage to the core of the VPBÉR-600 is less then 10^{-3} per reactor-year.

The design measures that have been taken and the qualitatively new level of safety eliminate the question of distance in the location of a power plant with a VPBÉR-600 reactor, make it unnecessary to evacuate the population, and allow the power plant to be located in the immediate vicinity of cities and other large energy consumers and used for the combined production of electrical energy and heat. The design and safety concept of the VPBÉR-600 characterize it as a new-generation installation of enhanced safety. The features of the integrated reactor, which prevent accident classes of large and moderate leakage, and the use of passively functioning and activated systems for emergency shut-down cooling determine its higher level of safety as compared with foreign installations of type AR-600. The problems associated with the development and creation of the equipment and the reactor as a whole have been solved.

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INTRODUCTION

The Project Summary Network shown on the attached Figures is typical for that of any large nuclear power project. The specific dates and the actual durations of individual activities are dependent on the overall contract schedule. Nevertheless, the activities shown and their relationship to other activities would, in general, be the same for any project. Each project will have unique characteristics depending on the country, its nuclear industry infrastructure, and the contractual arrangements for the project. Some projects have a requirement for significant localization of manufacturing, equipment supply, and site labor. Other projects are on a turnkey basis with no localization mandated. All such issues will result in unique characteristics of the project, including the actual project schedule.

For the overall project summary shown in Figure 1, the licensing portion of this figure should be ignored. This portion of the schedule was established for a specific country who has a unique licensing process.

The Project Summary schedule shown in Figure 1 has been developed specifically for the System 80+ Standard Plant Design. There are some unique features of the System 80+ design that are not applicable to other designs. These features, e.g., the incontainment refueling water storage tank, will result in slightly different relationships within the construction sequence. Nevertheless, the overall schedule is representative of any major nuclear power plant project.

Overall Summary Program

The overall summary program is presented in a format referred to as the Project Summary Network (PSN) (Figure 1, 3 sheets). The PSN is a schedule of activity at the total project level that displays the significant stages together with identification of key events and milestones against a calendar time scale.

The PSN divides the total project scope into the main work group elements. Major tasks within each work group are identified together with logic ties that indicate the significant relationships between specific tasks and events. These are: Licensing, Engineering & Procurement, Construction, Commissioning. The intervals between contract placement and start of site activities are clearly identified. The major constraints to the start of major areas of construction work, including major equipment deliveries, are shown. Whereas the summary program results in station operation in the year 2005 given the constraints imposed by first completing initial licensing and contract negotiation, the primary critical path through the project begins with site preparation. Early release of engineering and procurement for long lead items can result in a corresponding improvement in this schedule.

Since this schedule is at a summary level, it does not reflect the effect of different sites. Site differences would appear on schedules reflecting a higher level of detail. Such differences would not impact the overall project duration from authorization to commissioning.

Supporting Programs

The work that is described and scheduled in the Project Summary Network at a relatively high level is defined in greater detail in individual schedules for each major area. These programs are Level II and Level III Network Schedules. Each sublevel schedule breaks the activities down by primary work groups shown by the higher level schedules and shows the work in greater detail according to lower elements in a Work Breakdown Structure. The activity durations shown on the Level II correlate exactly with the corresponding summary activity durations, logic ties, and major milestones shown on the Project Summary Network. Level III relates work elements to components and resources, etc.

Design Programs

The major engineering work shown is reflected by the Engineering and Procurement Section on the PSN. As an example of a design program, at the next level, Figure 2 illustrates various design activities for the Nuplex 80+™ advanced control system and covers both software and hardware development, documentation and testing.

At the next level down, individual design activity would be further separated by phases of the design process, i.e., calculations, modelling, and drawing production, with preliminary and final issue dates, and specific vendor and station interfaces.

Ordering and Manufacturing Programs

A schedule showing the major NSSS and long lead equipment procurement, manufacturing, and delivery activities is shown on Figure 3.

The schedule is grouped by major component, and organized according to equipment type and purchase order. Key events in the procurement process, including specification development and required order dates are indicated. Following procurement, the manufacturing process is shown including dates for engineering and fabrication, fabrication and delivery lead times, and required at-site dates.

In sublevel schedules, key events are coded so that they can be related to corresponding events on other schedules, especially those for design and installation. The schedules are resource loaded in order to produce a table of the expected manpower levels required throughout the procurement and manufacturing process.

Construction, Installation, and Setting-To-Work

The Construction schedule is also shown in the PSN and has been developed to cover the construction work on all buildings and structures, including plant system and equipment installation, and the release and turnover points for each system. Key construction milestones are identified and coded so that they can be related to corresponding events on the Project Summary Network, and other Level II Schedules.

The schedule identifies basic temporary work and all required site services during construction. The schedule also identifies the start dates of all civil work, foundation pours, steel erection by building, building internals and setting to work by building. Start and finish dates are shown for piping systems, building services, electrical cable installation, controls and instrumentation installation. Dates for building enclosures, required services and terminal point releases for interfaces to work outside of scope are also be indicated on the schedule.

Construction has a duration of 54 months from first concrete to fuel loading. Once begun, the critical path is through the nuclear island concrete, containment sphere, primary loop components and the reactor shield building.

The Construction Schedule is labor intensive due to the short schedule duration. Two shifts have been assumed to reduce the manpower density. As usual for construction, civil work is followed by mechanical, electrical, and instrumentation and controls erection and installation. The construction organization will perform component tests as hydro tests prior to turning systems over to the commissioning organization.

Overall Commissioning

The Commissioning Program Schedule is reflected in Figure 4. The schedule shows all testing and key start-up activities. A system by system program with further breakdowns by sub-system where necessary would be developed as part of a detailed commissioning schedule and include the operator training programs. All activities are totally integrated both within a system and to other systems, and integrated with other final plant activities.

The schedule correlates with the turnover points indicated on the construction program as well as showing the major events and milestones reflected on higher level schedules.

Commissioning activities will begin approximately 36 months before fuel load. All system tests will be performed. The NSSS testing will include cold hydro and hot functional testing prior to fuel load.

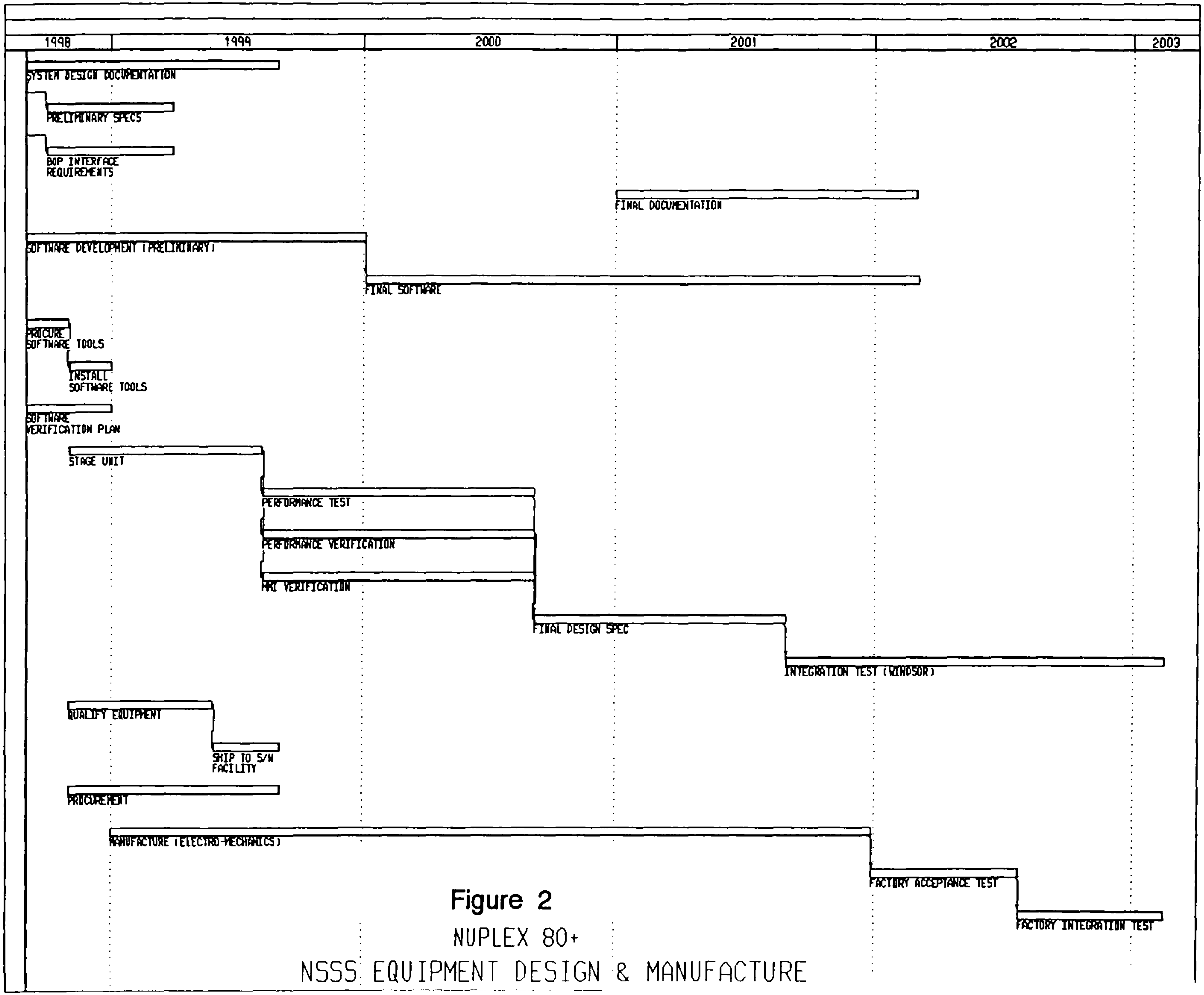

Figure 2

NUPLEX 80+

NSSS EQUIPMENT DESIGN & MANUFACTURE

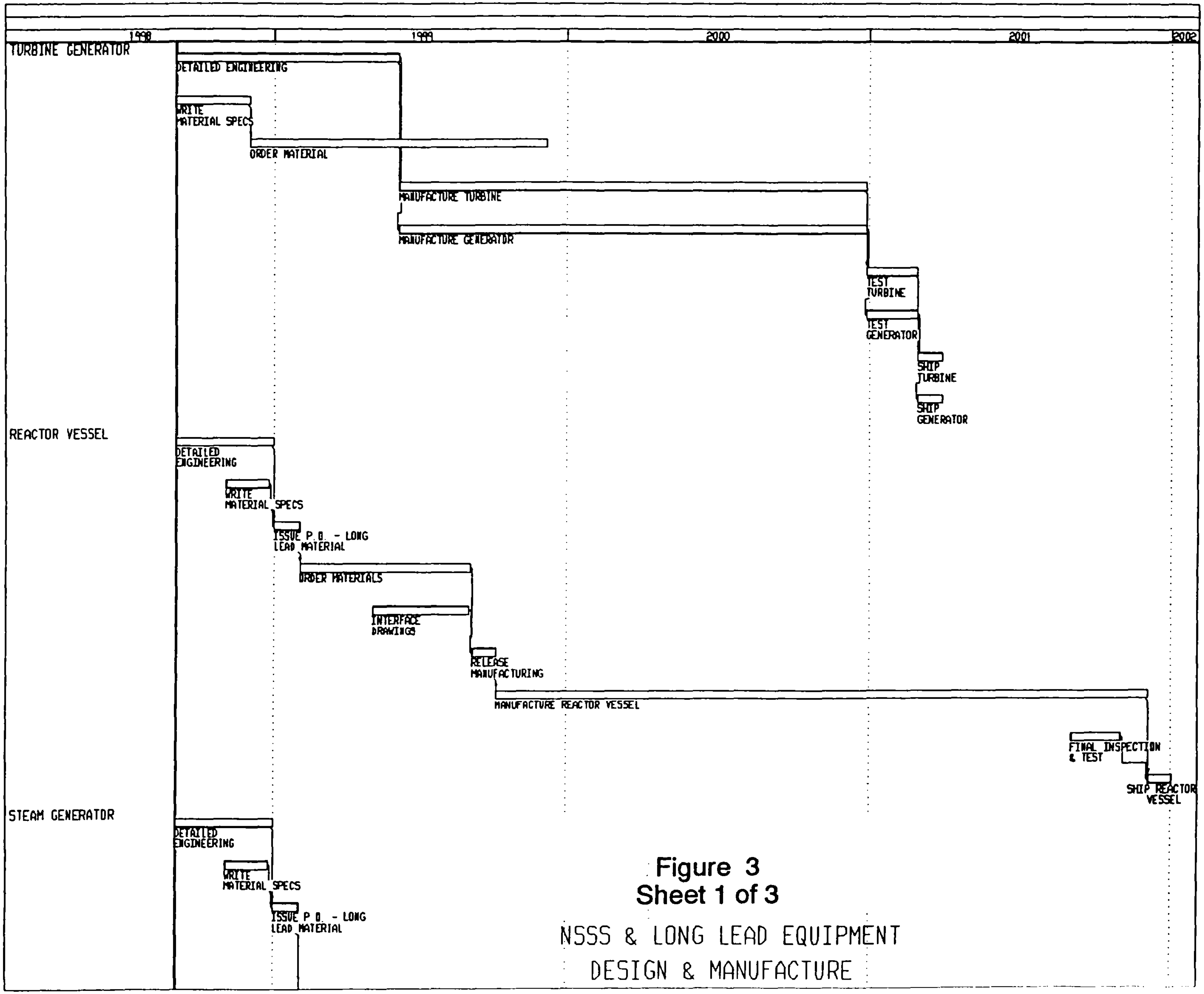

Figure 3
Sheet 1 of 3

NSSS & LONG LEAD EQUIPMENT
DESIGN & MANUFACTURE

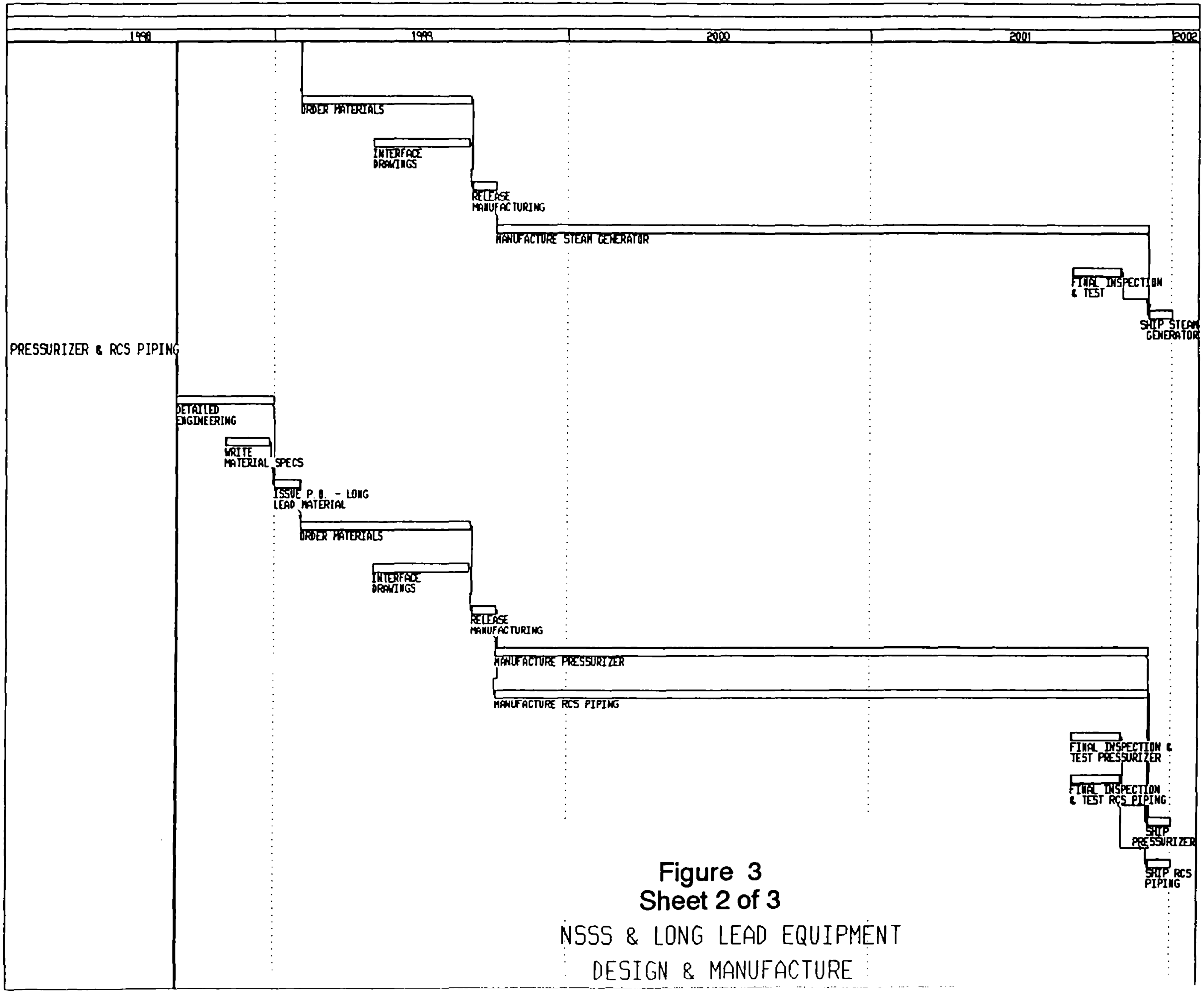

Figure 3
Sheet 2 of 3

NSSS & LONG LEAD EQUIPMENT
DESIGN & MANUFACTURE

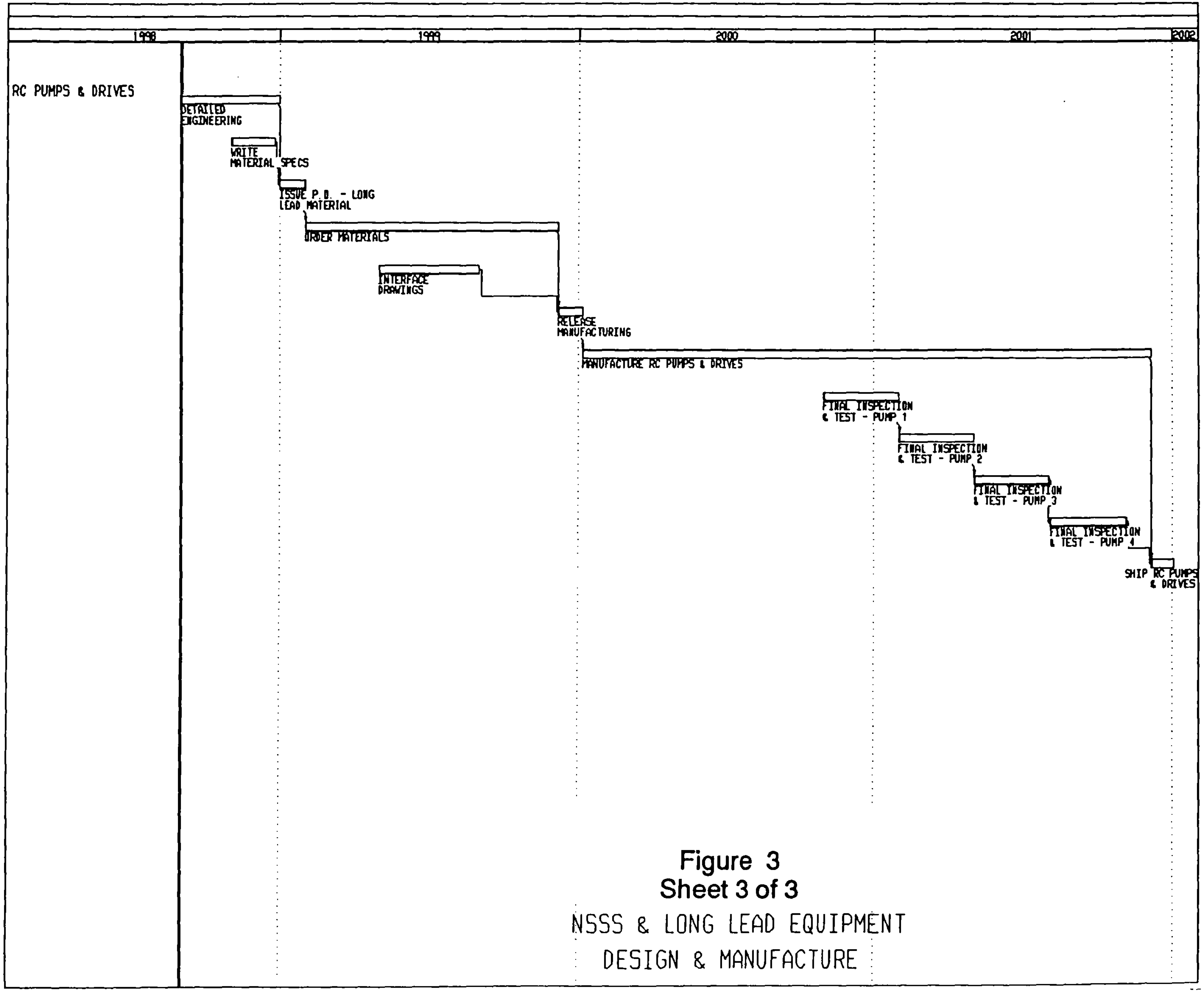

Figure 3
Sheet 3 of 3
NSSS & LONG LEAD EQUIPMENT
DESIGN & MANUFACTURE

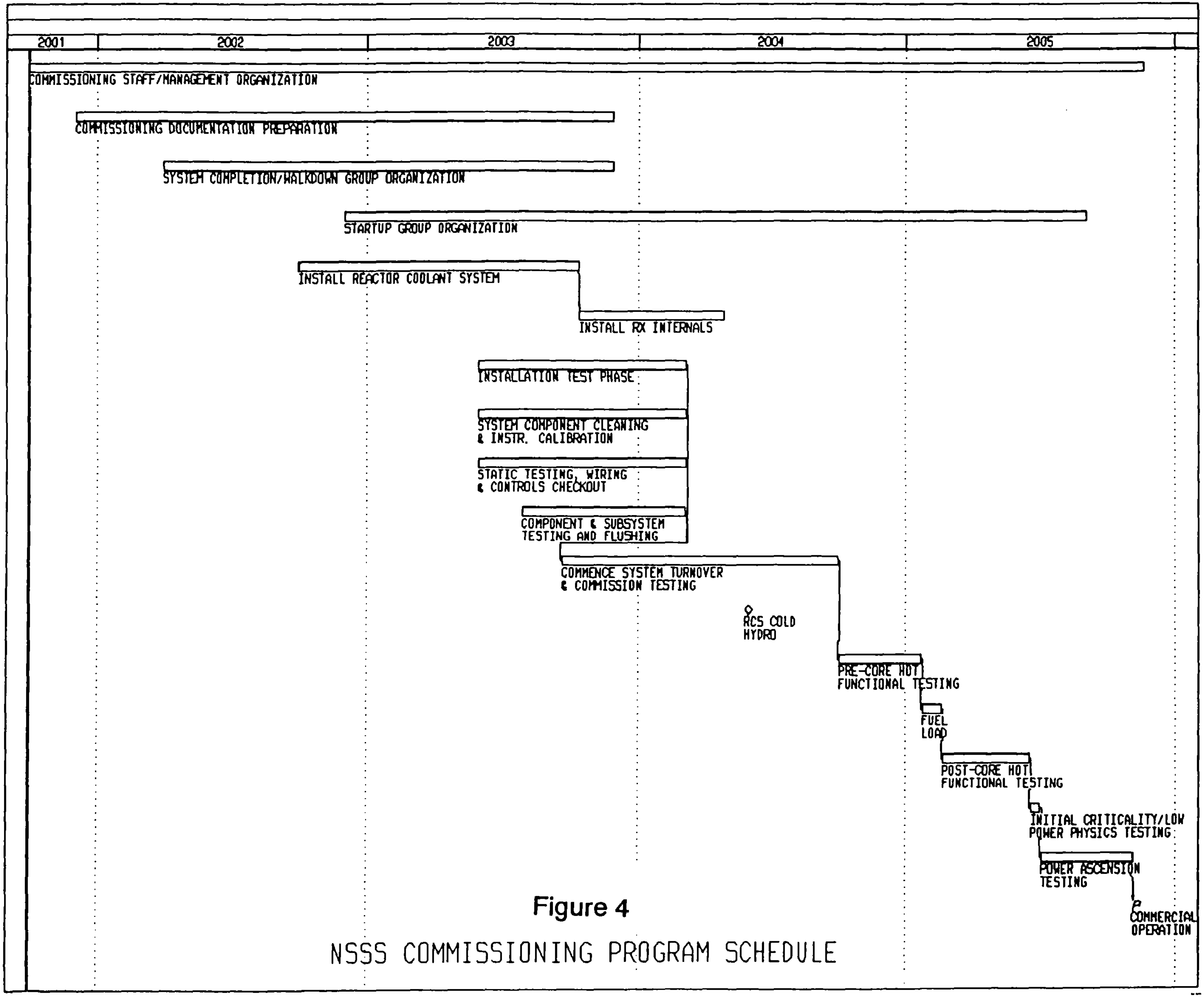

Figure 4

NSSS COMMISSIONING PROGRAM SCHEDULE

HB

5 MW of ELECTRIC POWER:

COST OF RESIDUAL OIL EQUIVALENT OVER 10 YEARS

(Estimates based on North Korean projections for 250 MW of electric power)

Installed Capacity	Power Generated	Heavy Oil Equivalent	COST- CIF 1/ Singapore LSWR at $87/mt FOB [LOW SULPHUR]	Singapore 380 at $73/mt FOB [HIGH SULPHUR]
5 MW	35 million KWh	10,000 metric tons (67,000 bbls)	$870,000 FOB $100,000 transp. $970,000	$730,000 FOB 100,000 transp. $830,000
		10 YEARS	x 10 $\boxed{\$9,700,000}$	x 10 $\boxed{\$8,300,000}$
250 MW	1,750 mil. KWh	500,000 metric tons (3,350,000 bbls)	$49,000,000	$42,000,000

1/ Delivered Cost from Singapore on Sept. 20, 1994; transportation cost $5-$10/metric ton (higher transp. cost used)

- note: Heavy Fuel Oil Costs this year volatile; for example, on Aug 12, 1994 cost of Singapore LSWR $127/mt and Singapore 380 $114/mt. 5 MW equivalent over 10 years for Singapore LSWR = $12.7 million; for Singapore 380 $11.4 million

COMPENSATION FOR ENERGY

(submitted by North Korea)

- It is considered that the substitute energy relevant to the electric output of 50 MW and 200 MW power plants is to be supplied.

- Electricity per annum

  . Output : 250 MW

  . Operating time : 7000 hours

  . Electricity : 250 MW x 7000 hrs =

  1,750,000 MWH

- Amount of heavy oil for electricity per annum

  . In case the heavy oil consumption per KWH is 290 g.

  290 g/KWH x 1,750,000 MWH = 500,000 t/annum ([illegible])
  1,000 [illegible] = 1 mil [illegible] = 1,934 [illegible] heavy oil or 290 mt (647 [illegible])

  . In case the heavy oil consumption per KWH is 300 g.

  300 g/KWH x 1,750,000 MWH = 520,000 t/annum

SUMMARY CONVERSION TABLE

A plant with capacity of:	**Can generate:**
1 Megawatt	8.760 x CF million kwh/yr where CF = capacity factor

Fuel required to generate:		**Requires:**
1 million kwh of electricity...		~~[illegible] heavy fuel or 280 metric tons~~ 1,744 bbl crude oil @ 5.93 million btu/bbl @ efficiency of 0.33
	or	277.3 cubic meters of oil @ 5.93 million btu/bbl @ efficiency of 0.33
	or	415.72 metric tons coal @ 24.87 million btu/bbl @ efficiency of 0.33
	or	2,605.3 million kcal @ efficiency of 0.33
1 million kwh/yr		4.778 bbl crude oil/day
	or	1,744 bbl crude oil/yr
	or	277.3 cubic meters oil/yr
	or	415.72 tons coal/yr

To convert from:	**To:**	**Multiply by:**
cubic meter oil	barrels oil	6.29
bbl oil	BTU	5.93 million
bbl oil	Kcal	1.494 million
ton of coal	BTU	24.87 million
ton of coal	Kcal	6.267 million
Kcal	BTU	3.968
bbl of oil	tons of coal	0.2384
cubic meter oil	tons of coal	1.500
tons of coal	bbl of oil	4.194
tons of coal	cubic meter oil	0.6668

Converting Power Plant Output to a Fuel Input Requirement:

Kwh/yr output to barrels or kiloliters of oil:

Two key factors effect the exact conversion factor:
1) The energy content of the oil (million BTU/bbl)
2) The conversion efficiency (EFF) of the power plant (fraction between 0 and 1, often near 0.33 for coal and oil plants)

Note that 1 kiloliter = 1 cubic meter = 6.29 bbl

Assuming oil with an energy content of 5.93 million btu/bbl and a power plant conversion efficiency of .33:

1 million kwh/year translates into:

```
1 million kwh/yr
     x 3412 BTU/kwh
          x (1/0.33)
= 10,340 million BTU/year
     x (1 bbl/5.93 million BTU)
= 1,744 bbl/year   or 261 metric tons/year (crude oil)
     x (1 year/365 days)
= 4.778 bbl/day
     x (1 kiloliter/6.29 bbl)
= 0.7596 kiloliter/day
```

If the efficiency is 0.35 instead, then multiply the above figures by .33/.35=0.9428. In other words, with a 35% efficient plant you need only 94.28% as much fuel as with a 33% efficient plant.

If the oil has an energy content of 5.80 million btu/bbl instead of 5.93 million btu/bbl, then multiply the above figures by 5.93/5.80=1.0224. In other words, with a lower energy content of the oil, we need 2.24% more barrels to supply the necessary energy.

Converting Power Plant Capacity to Output:

Power plant capacity is measured in Megawatts (MW)
Power plant output is measured in kilowatt hours per year (kwh/yr), or some power thereof.

No power plant can operate at 100% capacity all the time. It must come down for maintenance and repairs, and it may be throttled back during periods of slack demand. The capacity factor (CF) expresses the fraction of time (between 0 and 1.0) that the plant operates, on average.

1 MW capacity translates into:

1 MW
x (CF)
x 24 hrs/day
x 365 days/year
= 8760 Megawatt hours/year x CF
= 8760 Mwh/yr x CF
= 8760 thousand kwh/yr x CF
= 8.760 million kwh/yr x CF
= 0.00876 billion kwh/yr x CF

Example:

A 1000 MW plant operated at 80% capacity would generate how many million kwh/yr?

1000MW x 8.760 million kwh/yr x .8
= 7008 million kwh/yr

250MW = 1750 million kwh/yr